ANALYSIS OF THE INTEGRAL TRANSPORT THEORY CODE GELS AND ITS APPLICATION TO PWR CELL HOMOGENIZATION WITH THE PURPOSE OF THORIUM CONVERSION

by

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Ö Z E T

Bu çalışmada GELS kodu, alt programları ile birlikte detaylı olarak incelenmiştir. Bu kodu kullanarak yanma (burn up)lı veya yanma olmaksızın basınçlı su soğutmalı bir reaktörde (PWR) yakıt hücre hesabı ve kontrol çubuğu bölgesi hücre hesapları yapılarak böyle bir reaktörün çeşitli hücrelerinde Toryumdan Uranyum-233 üretim olanakları araştırılmıştır. Bu amaçla Uranyum bileşeni farklı zenginlikte değişik UO_2 - ThO_2 karışımlarını ihtiva eden yakıt hücrelerinin reaktivite değerleri ve , aynı zamanda bu hücrelerdeki özellikle Xenon-135 ve U-233 olmak üzere izotopların konsantrasyonları hem başlangıç değerleri hem de yanmanın bir fonksiyonu olarak ayrıntıları ile incelenmiştir.

A B S T R A C T

In the present work the computer code GELS has been examined in detail together with its subroutines. Using the GELS code fuel cell calculation, super cell calculation with and without burn-up were performed in a PWR.

The possibility of U-233 production from Thorium in a PWR is investigated. For this purpose fuel cells consisting of varying mixtures of UO_2 - ThO_2 with the Uranium component enriched to different levels are examined for their k_∞ values. Also, for these cells, both initially and as a function of burnup isotopic compositions, especially Xenon-135 and U-233 concentrations, are traced depending on exposure time.

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CHAPTER I

INTRODUCTION

The main object of this work is to investigate the possibility of Th conversion in a PWR of the type considered for the AKKUYU plant whose specifications are given in reference (1).

In the denatured Thorium fuel cycle, fuel is comprised of a mixture of Thorium and moderately enriched Uranium. During irradiation, Plutonium and U-233 will be produced from the fertile U-233 and Th-232 isotopes present in the fuel;

$$_{92}U^{238} + on^{1} \longrightarrow {}_{92}U^{239} \xrightarrow[\beta^-]{23.54\ \text{min.}} {}_{93}Np^{239} \xrightarrow[\beta^-]{2.33\ \text{days}} {}_{94}Pu^{239}$$

$$_{90}Th^{232} + on^{1} \longrightarrow {}_{90}Th^{233} \xrightarrow[\beta^-]{23.3\ \text{min.}} {}_{91}Pa^{233} \xrightarrow[\beta^-]{27.4\ \text{days}} {}_{92}U^{233}$$

U-233 is used in subsequent cycles to supplement the need for U-235 enrichment, thus reducing the enrichment cost. Presently the Plutonium produced from the fertile U-238 is not used in later cycles but stored for later use in fast breeder reactors. The reactors which use the mixture of Thorium and Uranium as fuel would be converters, producing energy and significant amounts of fissile isotope U-233 at the same time.

Thorium fuel cycle provides an alternate path toward a solution of the present energy requirements. The most significant advantage of Th - U^{233} cycle over the U^{238} - Pu^{239} cycle in thermal reactors is the potential of a higher conversion ratio (CR). In a converter or breder reactor CR units of fuel is produced for each unit of fuel consumed. The higher conversion ratio leads directly to a lower depletion charge in the fuel cycle cost.

So far, Thorium has not been used extensively in nuclear reactors but Th conversion may be feasible as a secondary aim in nuclear power production.

Advantages of the Thorium - Uranium-233 cycle are ;

1- U-233 has the excellent neutronic characteristics and deposits of Th in nature are known to be about two to three times as abundant as those of U.

2- It can be used for breeding fuel in thermal reactors

3- The melting point of Thorium (1842 ^{o}C) is higher than Uranium (1130 ^{o}C). UO_2 - ThO_2 rods in reactors can therefore be operated at temperatures of 1130 ^{o}C or greater, whereas Uranium rods are limited to 660 ^{o}C.

4- Uranium-233 is much less toxic than Plutonium.

5- The denatured Thorium fuel cycle has been suggested as an alternative which would allow the use ofbred fissile material while mitigating the potential for nuclear weapons proliferation present with Plutonium utilization. No nuclear fuel cycle can be made completely free of proliferation risk. The objective of proliferation resistant fuel cycles is to increase the diffuculty, cost, detectability and the time required to obtain a nuclear weapon so that nuclear power becomes an unattractive path to nuclear weapons.

In the denatured Thorium fuel cycle, the Uranium component of the fuel mixture would be enriched to $\leq$20 W/O U-235. The 20 W/O U-235 enrichment level is sufficiently low as to effectively preclude the construction of a nuclear weapon. The enrichment limitation of U-233 enriched Uranium is 12 W/O; the 12 W/O U-233 limit results in a similar unsuitability for weapons use as does 20 W/O U-235.[2]

6- The most advantageous mode of utilization of the Thorium fuel cycle is to recycle the generated U^{233}.

But the Th fuel cycle has a longer doubling time, which is one of the drawbacks of this cycle.

There are similarities and differences in the two fuel cycles; a major difference being the intermediate formation of Pa^{233} which is a strong neutron absorber, and decays slowly. Since the decay of Pa^{233} yields fissile U^{233}, reactivity control during shutdown is complicated. In the thermal-neutron spectrum, U^{233} has a much lower alpha (σ_c/σ_f) and therefore a higher eta [$\nu/(1+\alpha)$] than Pu^{239}. This difference in eta, however, is not as large as the difference in alpha might suggest, because Pu^{239} has a large ν or a larger number of neutrons produced per fission. In the thermal and epithermal energy regions, U^{233} has the potential for a higher conversion ratio than obtainable through the U-Pu fuel mixture when averaged over a burnup period. Therefore reactivity lifetimes can be larger and a greater return of fissile material at the end of the fuel cycle may be possible.

In Thorium systems the radioactivity of the daughter products of U^{232} in recovered Thorium and in U^{233} after the removal of fission products is a challenging problem. In less than a week after high-level decontamination, the gamma activity becomes sufficiently great so that fabrication by direct methods can be permitted only on a scheduled-radiation dosage basis.

The magnitude of this problem is directly related to the U^{232} concentration buildup that occurs throughout the exposure lifetime of the fuel. This, in turn, is a function of the integrated fuel exposure, including the neutron energy level incident upon the fuel materials since the principal reaction that produces U^{232}, the Th^{232} (n,2n) Th^{231} reaction, does not occur with neutrons with an energy below 6 Mev. The recovered fuel can be dealt with in several ways. The Thorium product can be stored for a long time to allow for the decay of Th^{228} (1.91 year half-life) or Uranium could be chemically purified just before the fabrication step.

This work constitutes the first part of a study investigating the possibility of U-233 production from Th^{232} as one of the aims in a dual purpose commercial PWR. It is assumed that the fuel in some regions of the core is a mixture of Uranium and Thorium dioxide. Initially the Uranium component of the fuel mixture would be enriched to $\leq$20 W/O U^{235}. The enrichment limitation of U-233 enriched Uranium is 12 W/O. The ratio of moderately enriched Uranium to Thorium in different fuel regions can be varied according to the nuclear and engineering specifications and adjusted to achieve the desired fuel burnup.

For this purpose, the integral transport code, GELS is used to prepare the inputs for different cells for the reactor analysis as a whole. GELS, is a one-dimensional, multigroup, integral transport theory code which produces few-group libraries for use in diffusion theory codes for pressurized light water reactor analysis. The whole range of possible temperatures is covered and the treatment of strong lumped absorbers as control or burnable poison pins is included.

Further, provision for burn-up is provided. The code has good accuracy and a fairly high computational speed. The code is furnished with a cross-section library of 15 fast and 30 thermal groups, suitable for general PWR applications. Those cross-sections being dependent directly upon fuel or moderator temperature are obtained for the actual case by square Lagrange interpolation from data resources prepared for several temparatures. Using the computer code GELS ; normal fuel cell calculation, control supercell calculation, burn-up fuel and supercell calculations can be made.

The PWR spectral code GELS consists of a MAIN program and several subroutines. First two subroutines reads only data from cards or tape. In six subroutines, transport calculation are performed.

MAIN program and all of the subroutines will be examined in great detail during the later chapters.

Analysis of the GELS code will be presented in chapter II. In chapter III, an application of this computer code will be presented and using the GELS cell calculations will be made for different UO_2-THO_2 fuel mixture in a PWR.

TABLE 1: - General Approximate Flow Diagram for GELS

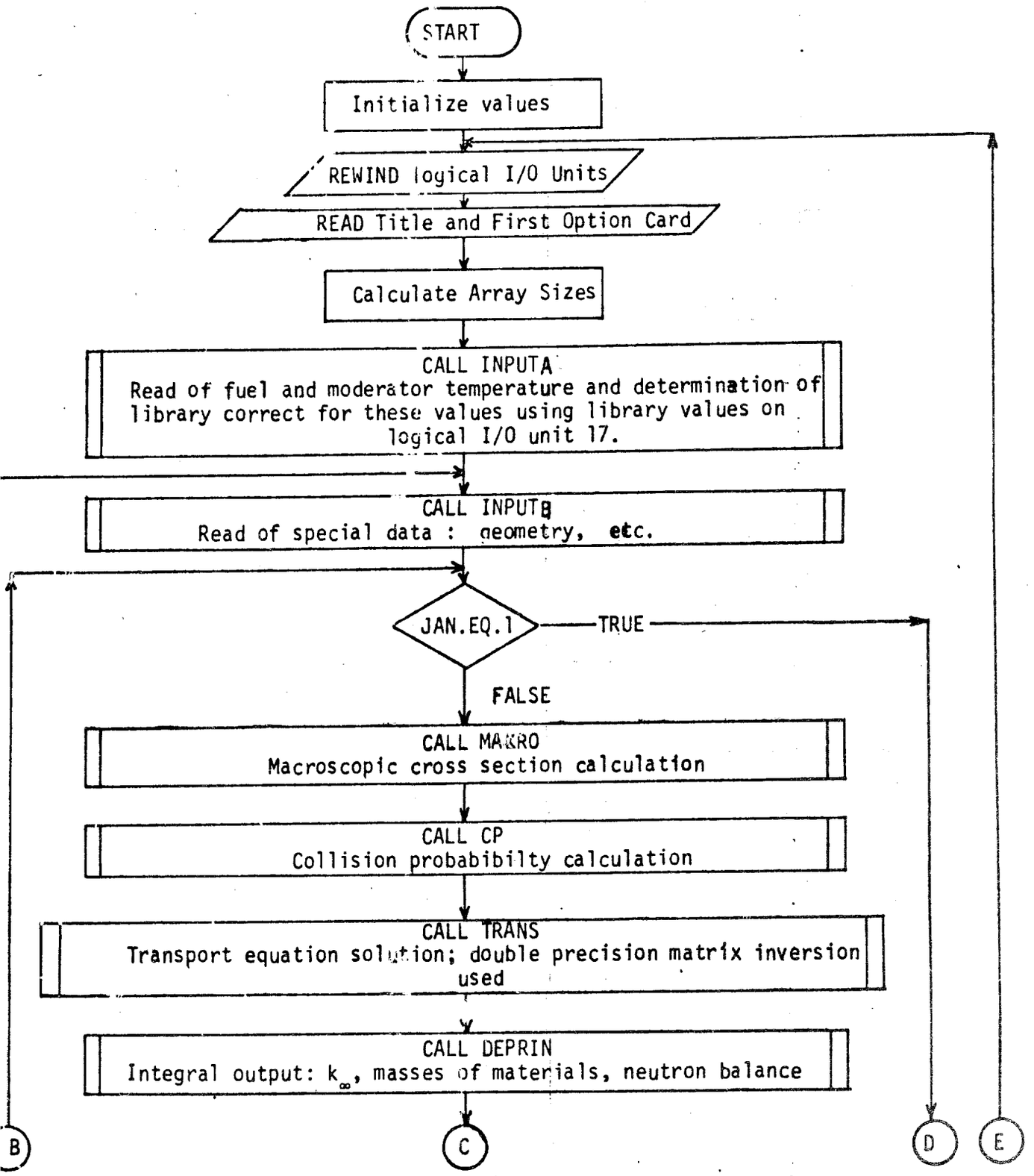


TABLE 1 - continued.

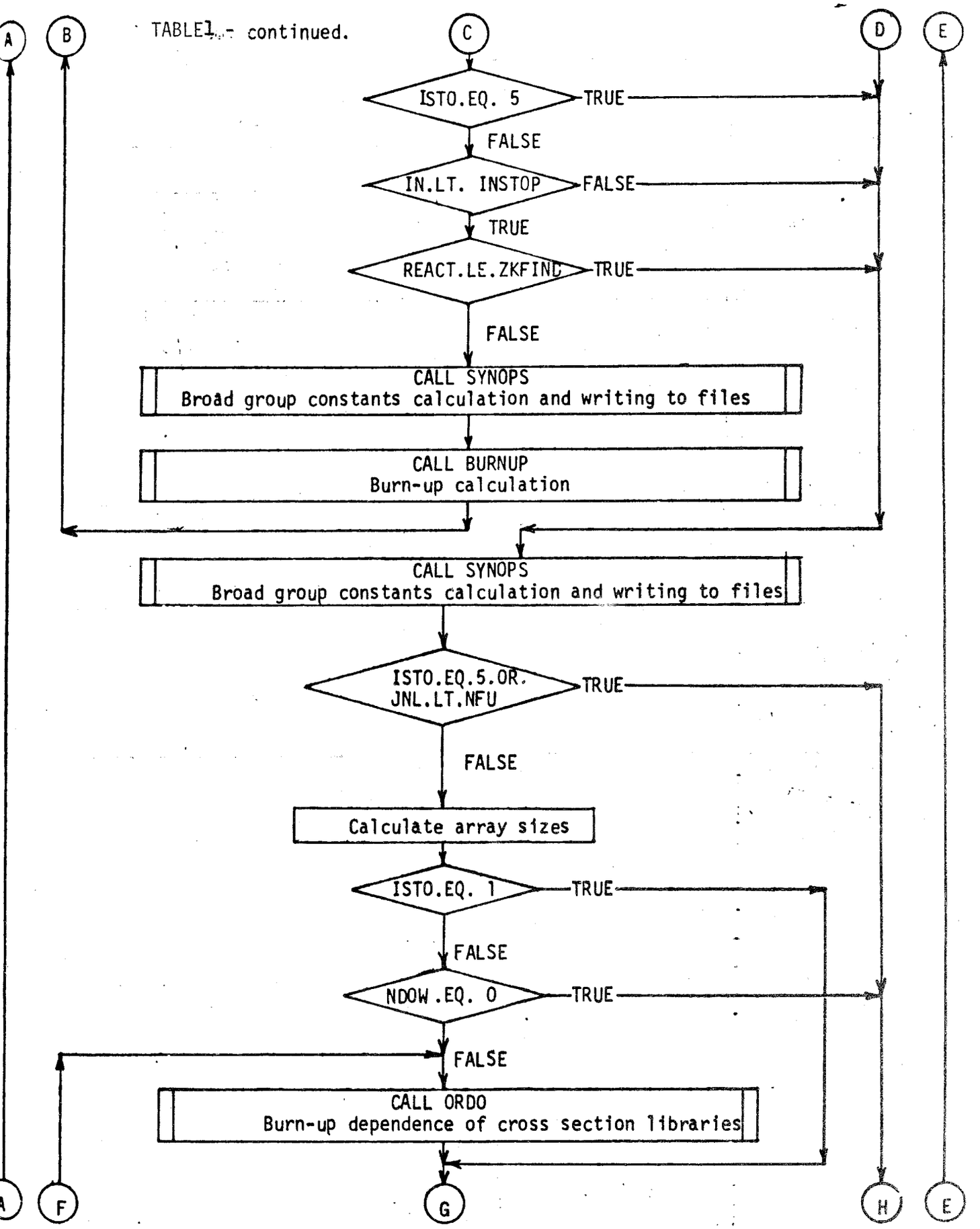


TABLE 1- continued.

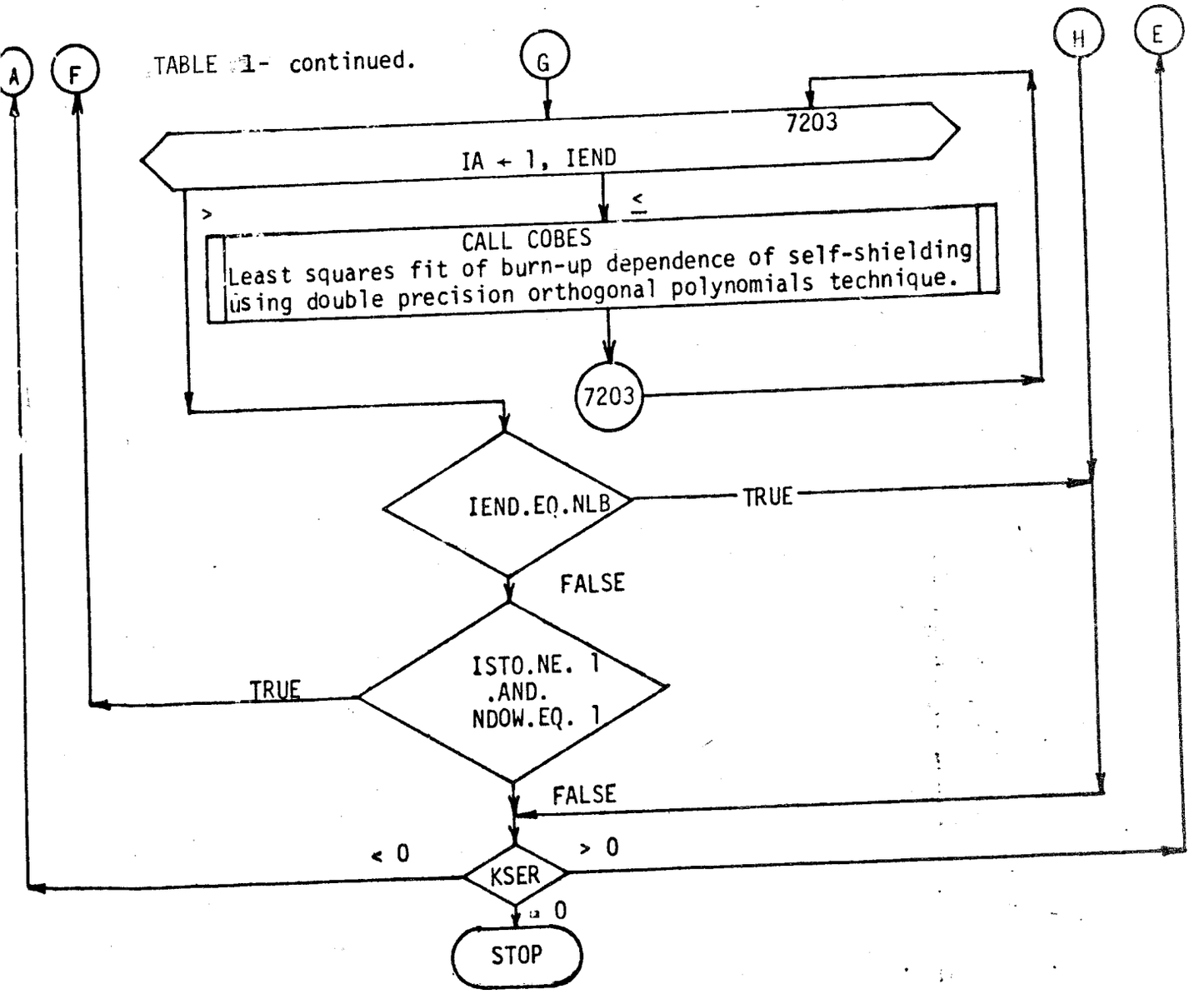
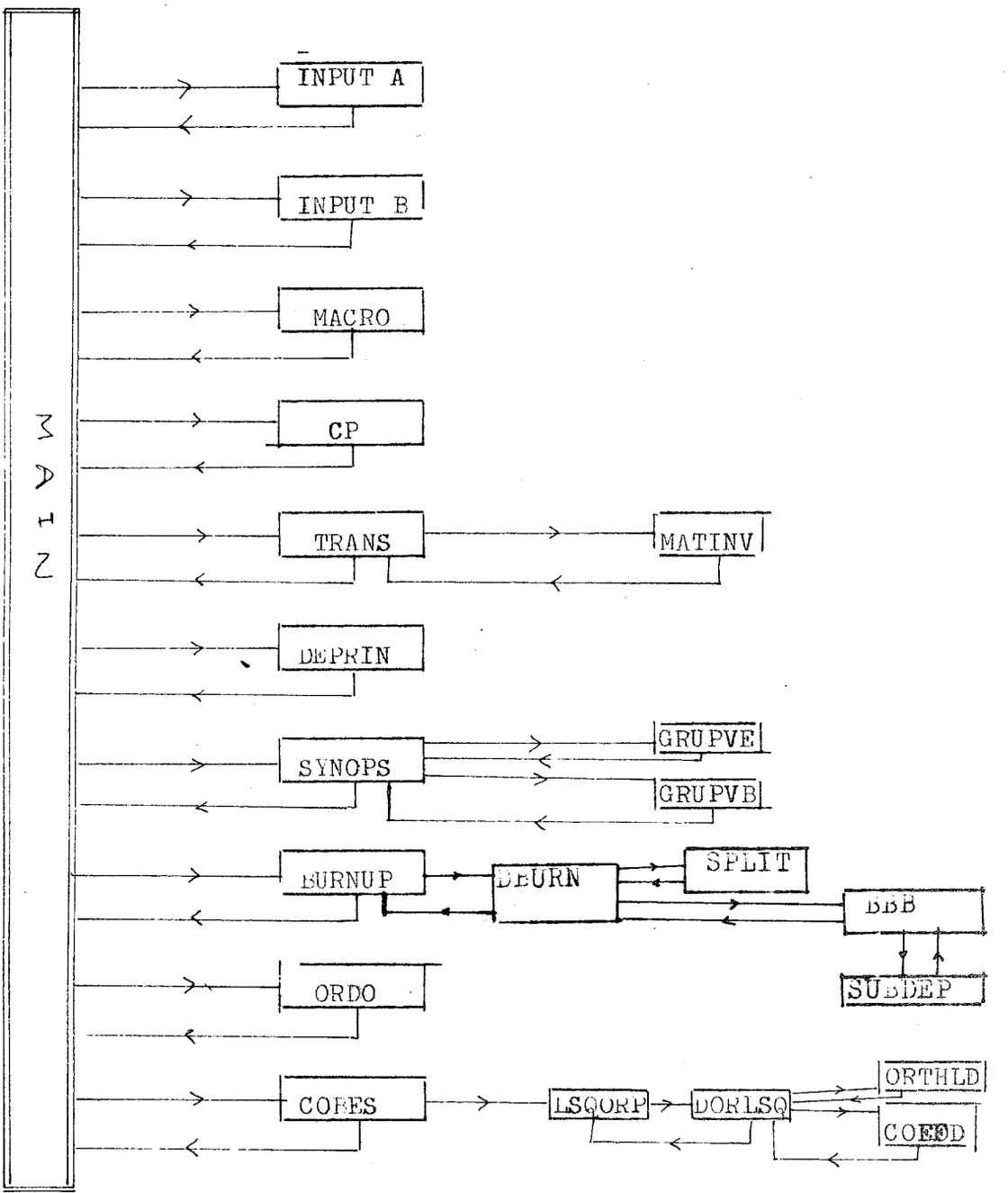


Table: 2 GELS

SIMPLIFIED FLOWCHART



CHAPTER II

ANALYSIS OF THE GELS CODE

II- 1. MAIN PROGRAM

The integral transport code GELS prepares the microscopic broad group cross sections and homogenized number densities for different cells within a reactor for use in diffusion theory codes making criticality searches. For this purpose each cell is divided into a number of regions. The densities of all isotopes, cell geometry and microscopic cross section library are given as input to this code via the subroutines INPUT A and INPUT B. Integral transport calculations are performed by using the collision probability method. The discretized form of the multigroup integral transport equation is given by

$$\sum_{T,j}^{g} \phi_j^g V_j = \sum_{i=1}^{N} P_{i,j}^g \cdot S_i \cdot V_i \qquad j = 1,\ldots\ldots N \text{ (number of regions)} \tag{2.1}$$

where the right hand side of this equation gives the total number of neutrons born in region j and that make their first collisions in region j. The left hand side gives the total interaction rate in region j. The total macroscopic cross sections appearing in this equation for all energy groups and regions are computed by the subroutine MACRO while the collision probabilities $P_{i,j}^g$ are calculated by the subroutine CP. The source term is calculated by making use of an initial flux guess. Then, the quantities obtained are inserted into the above equation which is then converted into matrix form. Finally the equation is solved for the fluxes by matrix inversion in subroutine TRANS.

As a result, the fluxes for each energy group and region are obtained. These fluxes are then normalized to specified power level in subroutine DEPRIN. Using the normalized fluxes, self shielding factors are calculated and all of the cross sections and number densities are homogenized in the subroutine SYNOPS so as to preserve the actual reaction rates. These homogenized number densities and normalized fluxes are then used in burn-up equations to obtain the time dependent changes in the concentrations of fuel isotopes and fission products in subroutine BURNUP. The MAIN program achieves this large task by making references to the subroutines mentioned above in the order of this presentation.

The theorical details of this qualitative outline will be presented in the analysis of related subroutines in the following chapters.

II- 2. TERMINOLOGY AND DEFINITIONS

ELEMENTARY CELL-ENLARGED CELL-SUPER CELL

In PWR's the non-cell water in the interassembly gaps can be added to the lattice dueto the following reasons ; the gaps are small in comparison with the thermal transport lenght of neutrons, these dissappear almost completely under operating conditions due to thermal expansion of the fuel assemblies. So, one can get a pure lattice at the reactor with a unique lattice constant (= enlarged pitch). A cell of this enlarged pitch is called an elementary cell.

Not every elementary cell has the same content. In order to achive a straight forward one-dimensional treatment of the whole micro structure, one can add one species of elementary cells to the cylindrized elementary cell of another species forming out of it one or more additional cylindrical regions.

If the species added does not have a peculiar structure of the neutronic flux density we call it an enlarged elementary cell. An example would be unrodded RCC-Cells added to fuel pin cells taking into account the numerical relation of both species.

If the species of elementary cells have a space dependent structure of neutronic flux density we call the whole a super-cell. For example rodded RCC-Cell or burnable poison pin cell surrounded by a number of fuel pin cells would constitute a supercell.

ZONE-MODERATOR ZONE-REGIONS

In the cylindrized (Super-) cell the term zone refers to every part which has a homogeneous material composition and which is bounded by parts of other material compositions. Due to cylindrization, the shape of every zone is either cylindrical or a cylindrical annulus. Every zone can be divided into a number of cylindrical annuli (of the same thickness within each zone). The elements of a such a subdivision we call regions.

All zones which contain the same moderator, ie. the same moderating isotopes of the same atomic densities are gathered to a "moderator zone". The non-moderating materials of zones belonging to the same moderator zone may be different. Zones without a moderator are neglected in the account of moderator zones.

In this version of GELS there is only one moderating isotope; the isotopic mixture H - 1/2 O. The only distinction between different moderator zones is the atomic density of this mixture.

II- 3. FUEL CELL AND SUPERCELL MODEL USED IN GELS CODE

Following calculations are performed by using the integral transport code, GELS :

Fuel cell calculation with and without burn-up,

Control supercell calculation,

Super cell calculation with burn-up

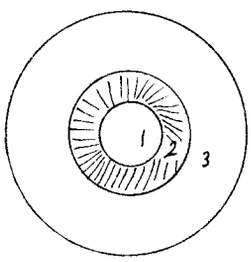
In the fuel cell calculation, it is assumed that the fuel is divided into four zones:

1 st zone: Fuel → gas gap

2 nd zone: Cladding

3 rd zone: Cell water → interelement water → spacers (inconel)

4 th zone: Cell water → interelement water → spacer → guide tubes (stainless steel)



1 Fuel pin

2 Zircalloy can

3 Moderator

Fig II- 1. Fuel pin cell

Poison "super" cell calculation:

1st zone: Ag-In-Cd absorber (usually divided into 4 regions)

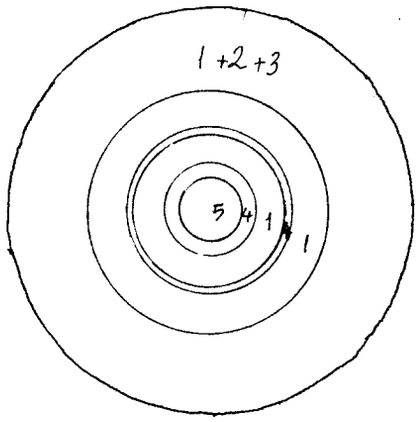
2nd zone: Stainless steel cladding → gas gap between rod and cladding

3rd zone: Water gap between cladding and guide tube.

4th zone: Stainless steel guide tubes.

5th zone: Cell water → interelement water → spacer (inconel) present (The radius of this zone is equal to the radius of the 3rd zone at the normal cell.)

6tn zone: Water → spacer → cladding → fuel



1 moderator

2 zircalloy can

3 fuel pin

4 stainless steel guide tube and control rod can

5 Ag - IN - Cd absorber

Figure (II-2): Super cell

Due to the one dimensional character of the code, the fuel assambly with RCC inserted has to be represented by a supercell.

In the case of supercell the GELS code provides for two different schemes of generating broad-group constants which differ in the homogenization procedure. In the first scheme, called burn-up supercell the set is produced for the supercell as a whole. And in the second case two sets are formed ; one for the cylindrized poison cell another one for the fuel/moderator mixture namely control supercell calculation. The group cross section library representing the fuel/moderator mixture will be formed by a run of the code homogenizing the fuel cell before the supercell calculation which follows as a second run.

During the fuel cell burn-up calculation the code produces a set of polynomial coefficients which correlate the self shielding or the disadvantage factors in each energy group with the U-235 concentration. These coefficients are then used in subsequent calculations of burn-up in poison cells namely in the burn-up supercell calculation.

II- 4. SUBROUTINE INPUT A

Input A is referenced from MAIN to read the data from cards or tape file so as to return to back to MAIN

In this subroutine microscopic cross sections are read from the file NUX or NUY for fuel isotopes, fission products, lumped absorbers, non burnable isotopes (structural materials) and moderator isotopes for 45 energy groups respectively.

Fission products are presumed to have absorption cross sections only. In control supercell case, the lumped absorber isotopes are not burnable, so that the cross sections of lumped absorber isotopes are hompgenized by the self shielding factors in the preceding case (i.e fuel cell calculation with ISTO = 4 option) and written into the NUY file.

Thus in the case of control supercell (ISTO= 5). the microscopic library belonging to lumped absorber isotopes are read together with non burnable isotopes from file NUY.

Transfer cross-sections are presumed to exist only for moderator isotopes.

Thorium-232, Uranium-238 and Plutonium-240 isotopes have resonance absorption cross sections. These cross sections are calculated by extrapolation depending on the fuel temperature.

This subroutine calculates the in-scatter cross sections for fuel isotopes, non burnable isotopes and lumped absorbers and the scattering cross section for the moderator isotope.

It also determines the fission yield of each fission product from the 6 fuel isotopes and prints the microscopic cross sections for fuel and moderator isotopes respectively and returns to the MAIN program.

This subroutine reads the following set of data from tape file NUX for all isotopes and 45 energy groups

In the case of ISTO = 5 NUX = NUY

CL 1 (IL), CL 2 (IL): Isotope name

TOSIG (IE, IL) : The microscopic transport cross section for isotope IL, for group IE

ABSIG (IE, IL) : The microscopic absorption cross section for isotope IL, and group IE

OUSIG (IE, 2,IL) : The microscopic out-scatter cross section for isotope IL and group IE

FISIG (IE, IL) : The microscopic nu fission cross section for IL th fuel isotope in IE th energy group

XNU (IE, IL) : The number of neutrons produced per fission for IL th fuel isotope for group IE

SIGFI (IE, L) = FISIG (IE,L) / XNU (IE,L) : The microscopic fission cross section for IL th fuel isotope for group IE

CHI (IE) : Spectrum for group IE

YIE (K,L) : The fission yield of K th fission product isotope from L th fuel isotope

OUSIG (IE, 1, IL) : Microscopic in-scatter cross section for isotope IL and group IE. This is found by subtracting the absorption and outscatter cross section from total cross section.

OUSIG (IE, 1, IL)= TUSIG (IE, IL) - ABSIG (IE, IL) - OUSIG (IE,IL)

OUSIGM (IE,IJ, ILM) : The microscopic transfer cross section for moderator isotope from group IE to group IJ

List of symbols used in this subroutine

IPOIS : Number of burnable isotopes

NBURB : Lumped absorber specification

N26 : Number of energy groups, N26 = 45

N23 : Number of fast energy groups, N23 = 15

NLB : Number of nuclides

NLM : Number of moderators

NLT : Number of fission products

N20 : Number of regions

NZM : Number of moderator zones

ISTO : Number of case type declaration, specified in MAIN

NUX : File name : NUX = 17

NUY : File name : NUY = 22

NFI : Number of fuel isotopes + number of fission products = 46

NOBB : Number of nonburnable isotopes

IE : Energy index IE = 1,....N26

IJ : Energy index IJ = 1,....N26

L : Isotope index L = 1,....NLB

IR : Region index IR = 1,....N20

II- 5. SUBROUTINE INPUT B

This subroutine is called from the MAIN program to read a set of data from cards and a set of polynomial coefficients from tape file 2 for burn-up supercell calculations. Recall that these polynomial coefficients are computed and written to file 2 during the fuel cell calculation (i.e ISTO = 1), and for burn-up supercell calculation (i.e ISTO = 2) these coefficients are read from file 2.

The macroscopic transfer cross section of moderator isotopes in each zone is computed as follows;

SIG (IE,IJ,IZ) = Q * OUSIGM (IE,IJ,ILM)

Where,

Q = DENM (ILM, IZ) ; Density of moderator in IZ th zone

The atomic density of Boron attachment to moderator in each region is calculated ;

DEN (NBOR, IR) = PPMS * DEN (NWA,IR)

PPMS = 1.01845 . $1.631978.10^{-7}$ * PPM is a convergence factor

The volume of each region and the total valume for the cell considered are computed as follows ;

VOL (IR) = π (RDR - RDRM) (RDR + RDRM)

$$VGE = \sum_{IR=1}^{N20} VOL(IR)$$

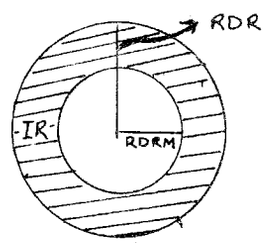


Figure (II-3): $VOL(IR) = \pi\left[(RDR)^2 - (RDRM)^2\right]$

Where, RDR = RD(IR) : the radius of IR th region

RDRM = RD(IR-1): the radius of (IR-1)th region

In the output, region, region radius, volume for region IR and total volume values are printed.

Finally the data depending on burn-up supercell calculation is written into the file NUS (12) at the end of the routine.

List of symbols used in this program:

IR : Region index

NBOR : Fuel isotopes + Fission products + 1

NWA = NLB Number of nuclides

DEN (L, IR) : Density of isotope L for region IR

DENM (IL, IZ) : Density of IL th moderator isotope for zone IZ

SIGS (IE,IJ,IZ): The macroscopic transfer cross section of moderator isotope from group IE to group IJ for zone IZ

Q = DENM (IL, IZ)

DEN (NBOR, IR): Density of Boron isotope for region IR

IZ : Moderator region index

NLBM = NLB - NLM : Number of nuclides except moderator isotopes

VOL (IR) : The volume of IR th region

VCE : Total volume of the cell

CO (K, IE): Polynomial coefficients

K = 6 Number of coefficients

IE : Number of energy groups

IL : Isotope index

II- 6. SUBROUTINE MACRO

It is called from the MAIN program to calculate the macroscopic cross sections a[illegible] returns to the MAIN.

At first, the program calculates the self shielding factors for burn-up supercell calculation, as will be explained in detail in the subroutine GRUPVE. If the case under consideration is not of burn-up supercell, the value of self shielding factors will be taken as unity. Using the self shielding factors, the program calculates the macroscopic cross sections such as ;

The macroscopic absorption cross section of all isotopes for group IE and region IR,

$$\sum a(IE,\ IR) = \sum_{L=1}^{NLB} N(L,\ IR) * \sigma a(IE,L) * SSP \qquad (2.2)$$

Where SSP is the self shielding factor.

For fuel isotopes macroscopic $\nu\Sigma f$, macroscopic in scatter cross sections and macroscopic out scatter cross sections are computed respectively as ;

$$\nu\sum f(IE,IR) = \sum_{L=1}^{NFUEL} N(L,IR) * \nu\sigma_f(IE,L) * SSP$$

$$\sum_s in(IE,IR) = \sum_{L=1}^{NFUEL} N(L,IR) * \sigma_{sin}(IE,L) * SSP \qquad (2.3)$$

$$\sum_s out(IE,IR) = \sum_{L=1}^{NFUEL} N(L,IR) * \sigma_{sout}(IE,L) * SSP$$

Macroscopic in scatter and out scatter cross sections for lumped absorber and non burnable isotopes are found as;

$$\sum s\,in\,(IE,IR) = \sum_{L=MI}^{NLBM} N(L,IR) * \sigma\, s\,in\,(IE,L) * SSP$$

$$\sum s\,out\,(IE,IR) = \sum_{L=MI}^{NLBM} N(L,IR) * \sigma\, s\,out\,(IE,L) * SSP$$

The total macroscopic cross section for group IE and region IR is ;

$$TOTL(IE,IR) = \sum_{IJ=1}^{N26} \Sigma_{s,t}(IE,IJ,IR) + \Sigma a(IE, IR) \qquad (2.4)$$

$$SIGD(IJ,IR) = \sum \sin(IE,IR) + \sum_{IE:IJ} \Sigma_{s,t}(IE,IJ,IR) \qquad (2.5)$$

Then, a return is made to **the MAIN program.**

List of symbols used in this program:

NFU= 6

MI = 46 : Fuel + fission products

JFI = 31 : Number of fission products

DIPCO (I,IE): Polynomial coefficient, I=1,..,6, IE = 1,..., N26(45)

DI = DEN(ICON,IR) : Atomic density of 5 th isotope for region IR

DENN(IE,IR) = $\sqrt{\ }^{g}$: The ratio of the average fuel to moderator flux. It is found in burn-up fuel cell calculation (ISTO = 1)

NLB : Number of nuclides

Q = DEN (L,IR) : The atomic density of L th isotope for region IR

SSP Self shielding factor

MOGI = NFUEL + NLT + Number of fuel and fission products

MOGIP = NFUEL + NLT + Lumped absorber

FISIG (IE,IL) : Microscopic nu* fission cross section of fuel isotope for IE th micro group.

TRIC (IE,IR) : Macroscopic nu* fission cross section of fuel isotopes for IE th micro group

SIGA (IE,IR) : Macroscopic obsorption cross section for group IE and region IR

OUSIG(IE, 1, L): Microscopic in-scatter cross section of L th isotope for IE th micro group.

TOUS (IE,1,LR) : Macroscopic in-scatter cross section for IE th micro group and IR th region

OUSIG (IE,2,L) : Microscopic out-scatter cross section of isotope L for IE th micro group

TOUS (IE,2,IR) : Macroscopic out-scatter cross section for IE th micro group and IR th region

IZ = IVC (IR) : Moderator region index

= 0 no moderator

IR = Region index $1 \leq IR \leq N20$

ASG = SIGS (IE,IJ,IZ) : The transfer cross section of moderator isotope from IE th group to IJ th group.

TOTL (IE,IR) : Total macroscopic cross section for IE th group and IR th region.

ALPHA : The fraction of the fuel volume to total volume of the cell

ALPHA = α : $V_f / V_f + V_m$

II- 7. SUBROUTINE CP

This subroutine is called from the MAIN to calculate the first collision probability matrices P^g_{ij}, the quantity P^g_{ij} is the probability that a neutron born in region i will have its first collision within the region j for energy group g.

Diffusion theory (or other PN approximation of low order) fails whenever the angular dependence of the flux is complicated. Instead of utilizing approximations of higher order in such situations, some special methods based on the use of collision probabilities in purely absorbing media are frequently useful.

Consider a common situation in which reactor fuel, localized in the form of lumps, e.g. rods, is surrounded by moderator. It is then sometimes useful to formulate the problem in terms of the probability that a neutron which appears in a region makes its next collision in that region. In a lattice structure, for example, fission neutrons may be born more-or-less uniformly in a fuel rod, then for the computation of the fast multiplication, it is required to determine the probability that these neutrons will undergo collisions in the rod before escaping. The neutrons which escape will be slowed down in the moderator and for resonance absorption calculation, the probability that the moderated neutrons will make their next collision in the fuel may be determined.

Collision probabilities have also been incorporated into widely used diffision theory calculations involving thermal neutrons.

In the typical one-speed collision probability calculation, the space is considered to be divided into a finite number of regions and it is assumed that neutrons are produced uniformly and isotropically in one of these regions. The problem is then to determine the probability that neutrons make their next collision in the source region or one of the other regions. The first collision probability matrices P^{g}_{ij} which bear the whole geometry of the problem are those of Carlvik.* Infinite circular cylinder has been taken for this calculation to represent the fuel cell.

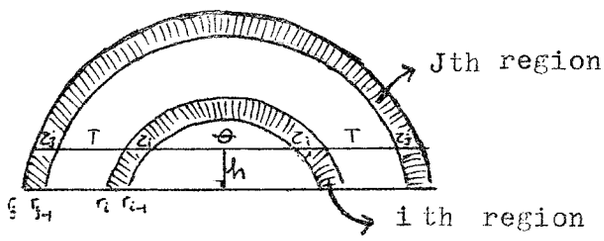


Figure (II-4)

τ_i : is the optical path in zone i

τ_j : the optical path in zone j

T : the optical path between zone i and zone j

θ : the optical path through the interior.

$$P_{ji} = \frac{1}{\Sigma_{t,j}} \frac{2}{\pi(r_j^2 - r_{j-1}^2)} \int_0^{r_{j-1}} dh \Big[K_{i3}(T) - K_{i3}(\tau_i + T) - K_{i3}(\tau_j + T) + K_{i3}(\tau_j + \tau_i + T) + K_{i3}(T + \tau_i + \theta) - K_{i3}(T + 2\tau_i + \theta) - K_{i3}(\tau_j + T + \tau_i + \theta) + K_{i3}(\tau_j + T + 2\tau_i + \theta) \Big]$$

Where,

$$K_{i,3} = \int_{-\pi/2}^{\pi/2} e^{-\frac{\pi}{\cos\theta}} \cos^2\theta \, d\theta \qquad (2.6)$$

3^{rd} order BICKLEY Function

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*) I. Carlvik, " A Method for Calculating Collision Probabilities in General Cylindrical Geometry etc." A/CONF. 28 P/681 (1964)

$$\tau_i(h) = \Sigma_{T_i}\left(\sqrt{r_i^2 - h^2} - \sqrt{r_{i-1}^2 - h^2}\right)$$

$$T(h) = \sum_{k=i+1}^{j-1} \tau_k$$

$$\theta(h) = \sum_{k:\,k_{min}}^{j-1} \tau_k \qquad r_{k_{min}} > h \qquad r_{k_{min-1}} < h$$

Collision probabilities P_{ij}^g for each energy group are computed through the use of the above equations.

II- 8. SUBROUTINE TRANS

This is called from the MAIN program to solve the multigroup transport equation and returns to MAIN.

The set of equations used to describe the transport problem is to the well known multi-group integral form, the two basic assumptions made are

- flat flux and source as computed by equation (2.8)
- isotropy of scattering as indicated in equation (2.9)

Reaction rates in the Jth region (j= 1,....N) are then described by;

$$\Sigma_{Tj}^g \, \phi_j^g \, V_j = \sum_{i=1}^{N} P_{ij}^g \, S_i \, V_i \qquad j = 1,...N \qquad (2.7)$$

Where

$$\phi_k^g = \int_{Vk} \phi^g(\vec{r}) \, d^3r \qquad (2.8)$$

$$S_k^g = \sum_{g'=1}^{G}\left[\frac{\chi^g}{k\infty}(\nu\Sigma_f)_k^{g'} + \Sigma_s^{g'-g}\right]\phi_k^{g'}$$

$$\Sigma_s^{g'-g} = 2\pi\int_{-1}^{+1}\Sigma_s^{g'-g}(\mu_o)\,d\mu_o \qquad (2.9)$$

Eq.(2.7) is solved as usual by replacement of the one eigenvalue problem (2.7) by G in homogeneous problems which continue to be coupled by an auter iteration mode. This is done by splitting from the source (2.8) the scattering rate belonging group g.

$$\Sigma_{Tj}^g \phi_j^g V_j = \sum_{i=1}^{N} P_{ij}^g S_i V_i = \sum_{i=1}^{N} P_{ij}^g \sum_{g'=1}^{G}\left[\frac{\chi^g}{k\infty}(\nu\Sigma_f)_k^{g'} + \Sigma_{s_i}^{g'\to g}\right]\phi_i^g V_i$$

$$j = 1,.....N$$

$$\left[\Sigma^{g}_{Tj} V_j - P^{g}_{ij} V_j \Sigma s_j^{g\to g}\right] \phi^{g}_{j} - \sum_{\substack{i=1\\ i\neq j}}^{N} P^{g}_{ij} V_i \Sigma^{g\to g}_{si} \phi^{g}_{i} = \sum_{i=1}^{N} \sum_{\substack{g'=1\\ g'\neq g}}^{G} P^{g}_{ij} \frac{\chi^{g}}{k_\infty} (\nu\Sigma_f)^{g'}_{i} \phi^{g'}_{i} V_i + \sum_{i=1}^{N} \sum_{\substack{g'=1\\ g'\neq g}}^{G} P^{g}_{ij} \Sigma^{g'\to g}_{si} \phi^{g'}_{i} V_i \qquad (2.10)$$

The in homogenous linear systems are solved by matrix inversion rather than by inner iterations for the fluxes.

Approximation (2.9) however, is more serious and a consistant P_1 representation could be worthwhile. Instead of this however, only a simple transport correction of the diagonal elements of the scattering matrix and, consequently, of the total macroscopic group constant is made,

$$\Sigma^{g}_{s1} = 2\pi \sum_{g'=1}^{G} \int_{-1}^{+1} \mu_0 \cdot \Sigma_s^{g'\to g}(\mu_0)\, d\mu_0 \qquad (2.11)$$

$$\Sigma^{g\to g}_{s0} \longrightarrow \Sigma^{g\to g}_{s0} - \Sigma^{g}_{s1} = 2\pi \int_{-1}^{+1} \Sigma_s^{g'\to g}(\mu_0)\, d\mu_0 - 2\pi \sum_{g'=1}^{G} \int_{-1}^{+1} \mu_0 \Sigma_s^{g'\to g}(\mu_0)\, d\mu_0$$

$$\Sigma^{g}_{T} \longrightarrow \Sigma^{g}_{a} + \Sigma^{g}_{s} - \Sigma^{g}_{s1}$$

Where Σ^{g}_{a} is the absorption and Σ_s is the scattering cross section.

Further, preferable data for diffusion calculations are generated in which the whole anisotropy has to be represented by the diffusion constant. Diffusion constant is calculated using Benoist's* formula

$$\bar{D}^{g} = \frac{\sum_{i=1}^{N} D_i^{g} \phi_i^{g} V_i}{\sum_{i=1}^{N} \phi_i^{g} V_i}$$

$$D_i^{g} = \frac{1}{3} \sum_{k=1}^{N} \frac{P^{g}_{ik}}{\Sigma^{g}_{Tk}} \qquad (2.12)$$

Which is monoenergetic but multiregional. By combination of (2.11) and (2.12) the reality is approximated at least on the level of diffusion theory.

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*) P. Benoist, " Theorie du Coefficient de Diffusion des Neutrons dans un Reseau Comportant des Cavites". CEA R-2278, 1964

This program solves multigroup transport equation by the following procedure:

Assuming that the flux , RPHI (IE,IR) = 1 , the total losses and productions are calculated for the cell considered

$$XLOSS = \sum_{IR:1}^{N:20} \sum_{IE:1}^{N:26} \Sigma a(IE,IR) * VOL(IR) * 1$$

$$XPROD = \sum_{IR:1}^{N:20} \sum_{IE:1}^{N:26} \upsilon \Sigma f(IE,IR).\ VOL(IR).\ 1\ ,$$

$$SUMTRI\ (IR) = \sum_{IE:1}^{N:26} \upsilon \Sigma f(I,IR) VOL(IR)$$

Multiplication factory k_∞ is then ;

REACT = XPROD / XLOSS

Using the value of $k\infty$, $\chi^g / k\infty$ is computed in order to find the updated source term on the right hand side of equation.

If there is moderator in region IRS and no moderator in region IRS respectively.

$$SUMSIG\ (IRS) = V\ (IRS).\sum_{IES:1}^{N:26} \Sigma_{s,trans}(IES,IE,IRS).\phi(IES,IRS).VOL(IRS) + \sum_{IES:1}^{N:26} \Sigma_{s,out}(IES,IRS).\phi(IES,IRS).VOL(IRS) \qquad (2.13)$$

$$SUMSIG\ (IRS) = \sum_{IES:1}^{N:26} \Sigma_{s,out}(IES,IRS)\ \phi\ (IES,IRS).VOL(IRS)$$

The left hand side and right hand side of equation can be converted into matrix form

$$C\ (IRS,IR) = \Sigma_T(IE,IR) - P(IRS,IE).\ SIGD(IE,IR).VOL(IR),\quad IR = IRS$$

$$C\ (IRS,IR) = -\ PIERS.\ SIGD\ (IE,IR).VOL(IR) \qquad IR \neq IRS$$

$$B\ (IRS) = \sum_{IR:1}^{N20} \left[SUMSIG\ (IR).PIERS; \chi^g/_{k\infty}\ .SUMTRI(JR) \right]$$

After this calculation the equation can be written as

$$[C]\ \vec{\phi}^g = \vec{B}$$

The subroutine MATINV is called to solve the matrix equation.

Using these fluxes, the new losses and production rates are calculated,

VLIR = VOL (IR)

RPHIX = (RPHI (IE,IR) -FEE(IR). V(LIR) (New flux-old flux) VOL(IR)

TRIX = TRIC (IE,IR). RPHI $\left[\nu \Sigma f(IE,IR)\ \phi(IE,IR).VOL(IR)\right]$

SUMTRI (IR) = $\sum_{IE:1}^{N26}$ TRIX $\left[\sum_{IE:1}^{N26} \upsilon \Sigma f(IE,IR)\ \phi(IE,IR).VOL(IR)\right]$

XPROD = $\sum_{IR:1}^{N20} \sum_{IE:1}^{N26} \upsilon \Sigma f(IE,IR)\ \phi(IE,IR).VOL(IR)\Big]$

XLOS = $\sum_{IR:1}^{N20} \sum_{IE:1}^{N26} \Sigma a(IE,IR)\ \phi(IE,IR)\ .\ VOL(IR)$

Calculates the new reactivity using these values

REACT = XPROD/XLOSS

Putting these values into eq(2.10) the same procedure is repeated for other groups. The multigroup transport equation (2.10) is solved for each energy group, therefore the above procedure is repeated for all energy groups in each outer iteration. Iterations are continued until the convergence criterion $\frac{k_\infty^{T+1} - k_\infty^{T}}{k_\infty^{T+1}} < 10^{-5}$ is satisfied in the MAIN program.

At the end of this subroutine, average diffusion coefficient is calculated for each energy group Eq(2.12)

VLIR = $\sum_{IR:1}^{N20}$ RPHI (IE,IR) .VOL(IR)

ADIF(IE,IR) = $\sum_{IRS:1}^{N20}$ P(IR,IRS,IE) VLIR/TOTL (IE,IRS)

SVLIR = $\sum_{IRS:1}^{N}$ VLIR DIF (IE) = $\sum_{IR:1}^{N20}$ ADIF

DIF (IE) = 0.333333333333333 .DIF(IE) / SVLIR

Finally a return is made to the MAIN program.

List of symbol used in this subroutine:

JN : Time step

VLIR = VOL(IR) : The volume of the region IR

XPROD : Production rate for all energy groups and regions in the cell

XLOSS : Total losses or absorption rate for all groups and regions in the cell.

RPHI (IE,IR): Denotes the flux for group IE and region IR

TRIC (IE,IR): Macroscopic nu* fission cross section for group IE and region IR $[\nu \Sigma f(IE,IR)]$

SIGA (IE,IR) : Macroscopic absorption cross section for group IE and region IR

TRIX = $\nu \Sigma f(IE,IR)$. VLIR

SUMTRI (IR) = $\sum_{IE=1}^{426} \nu \Sigma f(IE,IR)$. VLIR

REACT : Denotes the reactivity ($k\infty$)

IRS and IR: Region indices

IE and IES: Energy indices

ASG = SIGS (IES,IE,IZ) : The macroscopic transfer cross section of modarator isotope from group IES to group IE

PIERS = P (IR, IRS, IE) : Collision probability for group IE

C (IRS, IR) : Denotes the matrix on the left hand side of multi-group transport equation (2.10)

B (IRS) : Denotes the matrix on the right hand side of eq(2.10)

SIGD (IE,IR) : (Macroscopic transfer cross-section of moderator from group IE to IE+ macroscopic in scatter cross section of other isotopes for group IE) in region IR

ITERI : Iteration index

CONEIG = $\left| \frac{k_\infty^{T+1} - k_\infty^{T}}{k_\infty^{T+1}} \right|$ Convergence Criterion.

DIF (IE) : Average diffusion coefficient for group IE.

II- 9. SUBROUTINE MATINV

MATINV is called from the subroutine TRANS to solve the matrix equation using the matrix inversion with accompanying solution of linear equations.

Multigroup transport equation for each region was transformed to the matrix form by the subroutine TRANS

That is $[C]\ \vec{\phi}_j^{g} = \vec{B}_j \qquad j = 1,\ldots..N20$

Recall that

$$C(i,j) = \left[\sum T_j V_j - P_{ij} V_j \sum_{sj}^{g' \to g} \right] \quad \text{if } i = j \qquad \begin{bmatrix} i = 1, \ldots.N20 \\ j = 1,\ldots..N20 \end{bmatrix}$$

$$C(i,j) = \left[\rho_{ij}^{g} V_i \sum s_i^{g' \to g} \right] \quad \text{if } i \neq j \qquad (2.14)$$

$$B(j) = \sum_{i=1}^{N} \sum_{\substack{g'=1 \\ g' \neq g}}^{G} \rho_{ij}^{g} V_i \left[\left(\frac{\chi^{g}}{k_\infty} (\nu \Sigma_f)_i^{g'} + \sum s_i^{g' \to g} \right) \right] \phi_i^{g'}$$

The transport equation is solved for each energy group over all regions in the cell. So that the subroutine MATINV is referenced from TRANS for each energy group.

The subroutine MATINV inverts the matrix equation and finds the fluxes for each region in the cell, then returns to the TRANS to repeat the same procedure for the next energy group. This process is continued until the energy group is exceed N26 (45).

List of symbol used in this program:

ASP = C_i	the first term of the matrix equation
BSP = B_i	the right hand side of the matrix equation
NMAX = N20	number of regions
NSUB = N20	number of regions

II- 10. SUBROUTINE DEPRIN

DEPRIN is referenced from the MAIN to print the outputs and calculate the weights of heavy metals, macro group fluxes, fractional absorptions and productions. Also, it normalizes the fluxes to a given initial power level.

First items in the output to be printed are the time step; reactivity and atomic densities corresponding to the present time step for all isotopes. Then the weights of heavy metals (in grams) [i.e fuel and lumped absorber isotopes] are computed as follows;

For fuel isotopes,

$$GM(1, IR) = N(L,IR).\ VOL(IR).\ AWT(L)/0.60247 \qquad L=1.15,\ IR=1.N20$$

$$GS(L) = \sum_{IR=1}^{N20} N(L,IR).\ VOL(IR).\ AWT(L)/0.60247 \qquad (2.15)$$

$$FUEL = \sum_{L=1}^{NFUEL} GS(L)$$ which is the total weight of fuel isotopes in all regions.

Where, N(L,IR)th number density of isotope L for IR th region (in szilard), VOL (IR) : Volume of the IR th region.

AWT (L) : The atomic weight of the isotope L

0.60247 : Avagadro number in szilards.

For lumped absorbers:

$$GFM(L,IR) = N(L,IR).\ VOL(IR).\ AWT(L)/0.60247 \qquad (2.16)$$

$$GFS(L) = \sum_{IR=1}^{N20} N(L,IR).\ VOL(IR).\ AWT(L)/0.60247 \qquad L=47,..,49$$

All of the above quantities are printed in the output after these calculations.

On the other hand, to normalize the fluxes and to calculate the local form factor following calculations are made in the sequence shown,

First, DENOM is given by

$$DENOM = \sum_{IR=1}^{N26}\sum_{IE=1}^{N20}\sum_{L=1}^{NFUEL} N(L,IR).\ \nu f(IE,L).\ \phi(IE,IR).VOL(IR).\ SSP \qquad (2.17)$$

Where the summations are over all energy groups, overall regions and fuel isotopes and recall that SSP is equalto 1 except for the burn-up supercell calculations. For burn-up supercell calculation;

SSP= DENN(IE,IR)/ (1+ ALPHA.DENN(IE,IR) - 1)

Then the normalization factor is found by using the DENOM

$$FUNC = FIWATT.\ POWER.\ 10^{-24}/DENOM$$

Where FIWATT is given in MAIN as $1/\gamma = 3.125.\ 10^{10}$

POWER is the integral power per cm in watts, i.e power generation per unit lenght of fuel element., it is most commonly used when the fuel elements are cylindrical rods,units are in watt/cm.

Recall that, normalization factor= $\frac{1}{\gamma}.\ \frac{\text{Power(Watt)}}{\text{Fission rate}}$

Then all of the fluxes are normalized with this normalization factor;

RPHI (IE,IR) = FUNC. RPHI (IE,IR)

IE= 1,...N26

IR= 1,...N20

However to calculate the local form factor F(IR); the fission rate of the fuel isotopes over all energy groups in region IR and the average fission rate for the cell are computed in the following way

$$F_1(IR) = \sum_{L=1}^{NFUEL}\sum_{IE=1}^{N26} N(L,IR).\ \nu f(IE,L).\ \phi(IE,IR).\ SSP$$

$$AAA = \frac{\sum_{L=1}^{NFUEL}\sum_{IR=1}^{N20}\sum_{IE=1}^{N26} N(L,IR).\ \nu f(IE,L).\ \phi(IE,IR).\ SSP.\ VOL(IR)}{\sum_{IR=1}^{N20} VOL(IR)} \qquad (2.18)$$

And then local form factor is given by

$$F(IR) = \frac{F(IR)}{AAA}$$ (i.e, fission rate in region IR/avarage fission rate in the cell)

The program also calculates the macro group fluxes, those are fast and thermal group fluxes.

$$PHI(I,IR) = \sum_{IE=1}^{15} RPHI(IE,IR) \qquad IR = 1,\ldots N20 \qquad (2.19)$$

Which is the total fast group flux for each region, and,

$$PHI(2,IR) = \sum_{IE=16}^{49} RPHI(IE,IR) \qquad IR = 1,\ldots N20$$

Which is the total thermal group flux for each region. The results are printed in the output if the option NPRIN 1 = 0 holds. However, recall that if NPRIN 3 = 0, 45 group fluxes are printed and if NPRIN3 ≠ 0 2 group fluxes are printed. In the final section of the program, fractional absorptions and productions are computed as follows;

$$ABFRAC(IR) = \sum_{IE=1}^{N26} \Sigma_a(IE,IR) \cdot \phi(IE,IR) \qquad IR = 1,\ldots N20,\ IE = 1,\ldots N26$$

$$PRFAC(IR) = \sum_{IE=1}^{N26} \nu\Sigma_f(IE,IR) \cdot \phi(IE,IR) \qquad (2.20)$$

Total absorptions and productions in the cell are calculated as

$$ABSUM = \sum_{IR=1}^{N20} \sum_{IE=1}^{N26} \Sigma_a(IE,IR) \cdot \phi(IE,IR) \cdot VOL(IR)$$

$$PRSUM = \sum_{IR=1}^{N20} \sum_{IE=1}^{N26} \nu\Sigma_f(IE,IL) \cdot \phi(IE,IR) \cdot VOL(IR)$$

Total absorptions and productions for fast and thermal energy groups in the cell are calculated as;

$$QBAL(IE) = N(L,IR) \cdot \phi(IE,IR) \cdot VOL(IR) \cdot SSP$$

$$AB1 = \sum_{IR=1}^{N20} \sum_{IE=1}^{15} \overline{\Sigma_a}(IE,L) \cdot QBAL(IE) = \sum_{IR=1}^{N20} \sum_{IE=1}^{15} \overline{\Sigma_a}(IE,L) \cdot N(L,IR) \cdot \phi(IE,IR) \cdot VOL(IR) \cdot SSP$$

$$AB2 = \sum_{IR=1}^{N20} \sum_{IE=15}^{N26} \overline{\Sigma_a}(IE,L) \cdot QBAL(IE)$$

$$PR1 = \sum_{IR=1}^{N20} \sum_{IE=1}^{15} \nu\overline{\Sigma_f}(IE,L) \cdot N(L,IR) \cdot \phi(IE,IR) \cdot VOL(IR) \cdot SSP$$

$$PR2 = \sum_{IR=1}^{N20} \sum_{IE=15}^{N26} \nu\overline{\Sigma_f}(IE,L) \cdot N(L,IR) \cdot \phi(IE,IR) \cdot VOL(IR) \cdot SSP$$

$$AB1 = AB1/ABSUM = \frac{\sum_{IR=1}^{N20}\sum_{IE=1}^{15} \Sigma_a(IE,L).N(L,IR).\phi(IE,IR).VOL(IR).SSP}{\sum_{IR=1}^{N20}\sum_{IE=1}^{N26} \Sigma_a(IE,L).N(L,IR).\phi(IE,IR).VOL(IR).SSP}$$

$$AB2 = AB2/ABSUM = \frac{\sum_{IR=1}^{N20}\sum_{IE=15}^{N26} \Sigma_a(IE,L)N(L,IR).\phi(IE,IR).VOL(IR).SSP}{\sum_{IR=1}^{N20}\sum_{IE=1}^{N26} \Sigma_a(IE,L)N(L,IR).\phi(IE,IR).VOL(IR).SSP}$$

$$PR1 = PR1/PRSUM = \frac{\sum_{IR=1}^{N20}\sum_{IE=1}^{15} \nu\Sigma_f(IE,L)N(L,IR).\phi(IE,IR).VOL(IR).SSP}{\sum_{IR=1}^{N20}\sum_{IE=1}^{N26} \nu\Sigma_f(IE,L)N(L,IR).\phi(IE,IR).VOL(IR).SSP}$$

$$PR2 = PR2/PRSUM = \frac{\sum_{IR=1}^{N20}\sum_{IE=15}^{N26} \nu\Sigma_f(IE,L).N(L,IR).\phi(IE,IR).VOL(IR).SSP}{\sum_{IR=1}^{N20}\sum_{IE=1}^{N26} \nu\Sigma_f(IE,L).N(L,IR).\phi(IE,IR).VOL(IR).SSP}$$

Then

$ABT = AB1 + AB2$ $\quad QABT = \sum_{L=1}^{NLB} ABT$

$PRT = PR1 + PR2$

$QAB1 = \sum_{L=1}^{NLB} AB1$ $\quad QPR1 = \sum_{L=1}^{NFUEL} PR1$, $\quad QPR2 = \sum_{L=1}^{NFUEL} PR2$

$QAB2 = \sum_{L=1}^{NLB} AB2$ $\quad QPRT = \sum_{L=1}^{NFUEL} PRT$

Where

ABT represents the fractional absorption for each isotope and all energy groups.

RRT represents the fractional production for each isotope and all groups

QAB1 represents the fractional absorption for **the fast group and all isotopes.**

QAB2 represents the fractional absorption for **the thermal group and** all isotopes.

QABT represents the fractional absorption for all isotope and groups

QPRT represents the fractional production for all isotope and groups

Finally, printing these results in the output, a **return** is made to the MAIN program.

List of symbols used in this program:

L : isotope index L= 1,...NLB

VGE : total volume for the cell

N24 = N23 + 1 = 15 + 1 = 16

NPRIN 1 : Print index (output control parameter) specified in INPUT A

NLB : Number of isotopes

NFUEL : number of fuel isotopes

JN : number of time steps

REACT : reactivity for the cell considered

DEN (L,IR): atomic density of the Lth isotope for region IR

MOGI<L<MOGIP: represents the lumped absorbers

IR : region index IR=1,...N20

IE : group index IE= 1,...N26 (45)

SSP : self shielding factor

F (IR) : local form factor for region IR

RPHI (IE,IP): flux for group IE and region IR

PHI(1,IR) : fast group flux for region IR

PHI(2,IR) : thermal group flux for region IR

AWT (L) : atomic weight of isotope L

NPRIN 3 : print index

ABSUM : Total absorption rate over all regions and groups in the cell

PRSUM : total production rate over all regions and groups in the cell.

AB 1 :total absorption rate over all regions and over all fast groups in the cell IE= 1,....15

AB 2 : total absorption rate over all regions and over all thermal energy groups in the cell.

PR 1 : total production rate over all fast energy groups

PR 2 : total production rate over all regions and thermal energy groups.

AB1 = AB1/ABSUM : fractional absorption rate of isotope L for fast energy groups in the cell

AB2 = AB2/ABSUM : Fractional absorption of isotope L for thermal energy groups in the cell

PR 1 = PR1/PRSUM : fractional production rate of isotope L for fast energy groups in the cell.

PR 2 = PR2/PRSUM : fractional production rate of isotope L for thermal energy groups in the cell

ABT= AB1 + AB2 : Fractional absorption for fast group + fractional absorption for thermal group, for isotope L in the cell.

II- 11. SUBROUTINE BURNUP

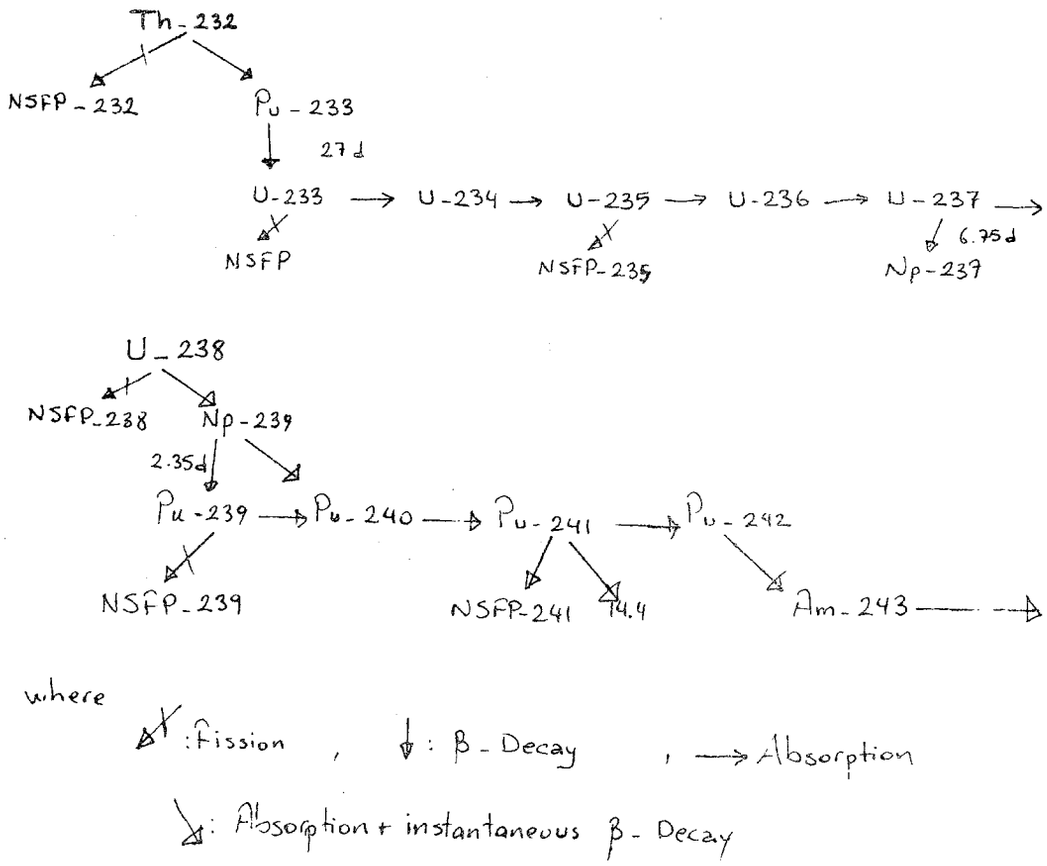
Burnup is referenced for each time step from the MAIN in order to solve the system of burnup equations and to find the number densities depending on the time step for all isotopes involved in the cell.

Burnup equations describe changes in the isotopic concentrations with time at a point or a region in the reactor core. The point or region is a unit of volume which is either homogenous in composition or has been properly homogenized by volume and flux weighting.

During the operation of a reactor, the fissile nuclides are consumed by fission and about two hundred different isotopes exist in the core after a sufficiently long period of time. Some nuclei are direct fission products and others are formed through radioactive decay. A number of those fission products have high or moderately high cross sections for neutron capture, consequently they have a significant influence on the neutron economy of the system. Furthermore, the conversion of fertile nuclei into fissile nuclei has of course, an important effect on reactor life time and control. In addition, radioactive capture of neutrons by both fissile and fertile species leads to the formation of such nuclei as Uranium-236, Plutonium-240, Uranium-239, and so on. These can also capture neutrons or suffer beta decay or both so that many new heavy isotopes, i.e, isotopes of Thorium, Protactinium, Uranium, Neptunium, Plutonium, etc. are present in the fuel after a period of reactor operation.

Cross section libraries pertinent to the isotopic chains involved are needed for burnup calculations. For the sake of convenience the avoidance of several pseudo fission products with different saturation behaviours was considered to be desirable. While the number of heavy metals is fixed in some measure by the appliciations planned, the number of fission products has to be optimized, so that it is as small as possible allowing at the same time for a good burnup calculation.

Morever, the structure of the chains should be simple. Fig (II.5) shows the heavy metal chains. The six important isotopes are equipped with a non-saturating pseudo fission product representing the low **a**ross section absorbers,



NSFP : Non saturating fission product

Figure (II-5) : Heavy metal chains

The assumptions involved in abbreviating the chains are the following:

1. The yields of precursors with half lifes below the order of magnitude of hours are cumulated to that of the next longer living successor. In a way those precursors are presumed to decay instantly to the longer living successor isotope.

2. Low yield nuclides (yield, $\leq$ 1%) without strong fission product parents are omitted even if their absorption cross sections are high.

3. Unstable nuclides of a sufficient lifetime are taken into account, even if their absorption is small, with regard to burn-up effects caused by power cycling.

This special choice of fission products facilitates the solution of the system of burn-up equations ;

$$\frac{d\,N_i(t)}{d\,t} = C\ N_i^T(t) \qquad (2.21)$$

$$N(t) = \left\{ N_1(t) \ldots\ldots N_{nf}(t),\ N_{nf+1}(t) \ldots\ldots N_n(t) \right\}$$

Where,

n : number of burnable nuclei

nf : number of fissionable nuclei

np : number of fission product

N_i : density of nuclei

T : transposition symbol

C : Depletion matrix

C being of lower triangular form has been brought to a more diagonal shape, by eliminating the nf. np block of elements which couples the fission products with the fissionable nuclei. Elimination can be performed by assuming the fission products not to originate continuously from the fissionable nuclei but from an external source term of the form:

$$\gamma_k = \sum_{i=1}^{nf} y^{i\to k}\, \bar{N}_i\, P_i$$

$$\bar{N}_i = \frac{1}{t-t_0} \int_{t_0}^{t} N_i(\tau)\, d\tau$$

$$P_i = \sum_{g=1}^{G} \nu \Sigma_{f,i}^{g}\, \phi^g \qquad (2.22)$$

$$k = n_{g+1}, \ldots\ldots n_{f+np}$$

Where

$Y^{i\to k}$: yield of fission product k from fissile isotope i

G= 45 : number of energy groups

t-to : lenght of time step

P_i : specific neutron production rate per isotope i

This approximation is justified by the low share of fission products in the total neutron absorption balance. Moreover the time step lenght t-to, is limited by the basic assumption that the regional flux in eq.(2.22) is constant. Eq (2.21) holds for every spatial region seperately.

Approximation (2.22) turns the homogeneous system (2.21) into an inhomogeneous one but the new depletion matrix has only three generally non-vanishing successive elements per row:

$$C = \left\{ C_{i,i-2} \quad C_{i,i-1} \quad C_{ii} \right\}$$

Under the above considerations depletion equations at a region have the form:

$$\frac{dN_i}{dt} = -N_i\left(\lambda_i - \sum_{g=1}^{G} \nabla a_i^g \bar{\phi}^g\right) + N_{i-1}\left(\lambda_{i-1} + \sum_{g=1}^{G} \nabla c_{i-1}^g \bar{\phi}^g\right) + N_{i-2}\sum_{g=1}^{G} \nabla_{c_{i-2}}^g \bar{\phi}^g \qquad (2.23)$$

Where;

N_i : the number density (nuclei/cm^3) of the isotope i

λ_i, λ_{i-1} : decay constant for the isotope i and i-1

$\bar{\phi}^g$: average, power-normalized flux for energy group g

∇c_{i-1}^g, ∇c_{i-2}^g : capture cross sections for energy group g and isotopes (i-1) , (i-2)

∇a_i^g : absorption cross section for group g and isotope i

G = 45 : number of energy groups.

Then the elements of the depletion matrix C becomes:

$$C_{ii} = \lambda_i + a_i$$

$$C_{i,i-1} = \delta_{i,1}(a_{i-1} - \beta_{i-1}) + \lambda_{i-1} \qquad (2.24)$$

$$C_{i,i-2} = \delta_{i,2}(a_{i-2} - \beta_{i-2})$$

$$i = 1, \ldots, n$$

$$a_i = \sum_{g=1}^{G} \nabla a_i^g \bar{\phi}^g$$

$$\beta_i = \sum_{g=1}^{G} \nabla \beta_i^g \bar{\phi}^g$$

$$\delta_{i,1}; \delta_{i,2} = \begin{cases} 0 \\ 1 \end{cases}$$

coupling indicators, actual value due to chain structure

In the case of a pseudo fission product the complete row vanishes

$$C_{i,i-2} = C_{i,i-1} = C_{i,i} = 0$$

By equations (2.22) and (2.23) the solution of equation (2.21) has been simplified, that index calculations become unnecessary and the recurrence relations for the coefficients of the solving sum are quite short.

The change of nuclide i under burnup is decribed by:

$$N_i(t) = \sum_{j=iA}^{i} a_{ij} e^{-c_{ij}(t-t_0)} + b_i$$

$$a_{i,j} = \frac{c_{i,i-1} a_{i-1,j} + c_{i,i-2} a_{i-2,j}}{c_{ii} - c_{jj}}$$

$$b_i = \frac{c_{i,i-2} b_{i-2} + c_{i,i-1} b_{i-1} + \gamma_i}{c_{ii}} \qquad (2.25)$$

$$a_{ii} = N_i(t_0) - \sum_{j=iA}^{i-1} a_{ij} - b_i$$

$$a_{kl} = 0 \quad \text{if} \quad l > k$$

$$b_i = 0 \quad \text{if} \quad i \leq n_f \text{ or } i > n_f + n_p \,, \quad iA \leq 1$$

Where iA: index of the first element in the chain,

In the case of pseudo fission products eq (2.25) degenerates into:

$$N_{fp}(t) = N_{fp}(t_0) + \gamma_{fp}(t - t_0) \qquad (2.26)$$

The subroutine burnup normalizes the fluxes and calculates absorption, fission and capture rates for each isotope in a region. Also decay rates are found for the most important nuclei. Then the lower triangular depletion matrix is established and split into a number of independent fuel and fission product chains. The coefficient matrix occuring in equation (2.25) is computed and using these values set of equations of the form (2.25) are solved. Consequently, the number densities for fuel and fission products are obtained for the present time step. Then a return is made to the MAIN program.

II- 12. SUBROUTINE SYNOPS

SYNOPS is called 3 times independently from the MAIN according to the value of the option NSYN to perform the cell homogenization and to calculate the microscopic and macroscopic broad group constants. The option NSYN has three possible values specified in MAIN i.e, NSYN= -1,0,1 . If NSYN is equal to 1, SYNOPS is referenced from MAIN to print the results of burn-up calculations for each time step. When NSYN is equal to 0 or -1, SYNOPS calculates the broad group constants and writes them to file NUY and IFREB with the help of its sub-programs GRUPVE and GRUPVB. Then the program calls subroutine GRUBVE to perform the cell homogenization and calculates the self shielding factors. A variety of methods have been developed to account for the heterogenity in a reactor lattice cell. Such homogenization techniques are used for calculating the physical parameters for a unit cell that may be defined as a single fuel rod together with its associated cladding gap and moderator. Fuel, water and structural materials are assumed to be homogenized over the volume of the unit cell, and the characteristics of the cell are computed for the homogeneous mixture. An important problem associated with homogenization is the difference in flux levels in different materials within the cell.in a typical water-reactor lattice. The homogenized area is shown in figure (II-6).

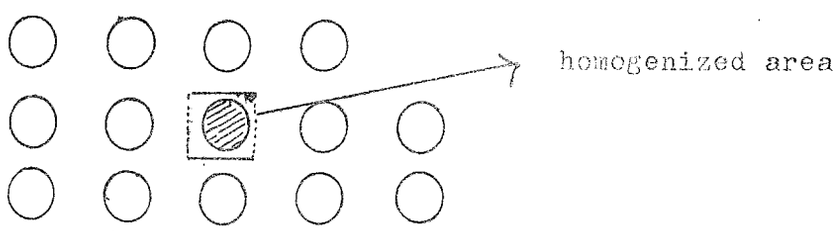


Figure (II-6)

When the cell is homogenized the effective cross sections are defined in such a way that the reaction rates are preserved when integrated over a cell. Suppose, for example, that $\nabla_x(r)$ represents the cross section for a given reaction x, for neutrons of a given energy at position r within the cell. If $\phi(r)$ is the computed flux in the cell calculation, then the effective cross section $\overline{\nabla_x}$, may be defined as,

$$\overline{\nabla_x} = \frac{\int_{cell} \nabla x(r)\ \phi(r).dV}{\int_{cell} \phi(r).dV} \qquad (2.27)$$

Shielding (or disadvantage) factors can also be defined for any kind of neutron reaction as the ratio between the actual reaction rate and that which would be found for the same material exposed to the volume averaged flux, thus, the shielding factor, S_x for a reaction of type x can be represented by,

$$S_x = \frac{\int_{cell} \nabla_x \phi\, dV}{\frac{\int_{cell} \phi dV}{V_{cell}} \int_{cell} \nabla_x . dV} \qquad (2.28)$$

In terms of S_x, therefore, $\overline{\nabla_x}$ may be written as

$$\overline{\nabla_x} = S_x \frac{\int_{cell} \nabla_x dV}{V_{cell}}$$

By using the effective cross sections or shielding factors, all reaction rates in the homogenized cell will be equal to those in the actual heteregeneous cell.

Using the subroutine GRUBVE, SYNOPS performes the homogenization and calculates the microscopic broad group constants for the cell under consideration. After this procedure is completed, SYNOPS also calls the subroutine GRUPVB to calculate the macroscopic broad constants and writes input to EREB file for use in the EREBUS code.

If the burnup fuel cell calculation is to be considered (i.e ISIO=1) SYNOPS does not call the subroutine GRUPVB. Because in this case it only prepares the coefficients of the polynomial for use in the burn-up super cell calculation (i.e, ISTO = 2).

Finally, the program writes the weights of heavy metals and the reactivity to file NU(13) after which a return is made to the MAIN program. On the other hand, if SYNOPS is called from MAIN with the option NSYN = 1, the program prints as the output; time step (IN), time (days), weights of heavy metals (grams) and burn-up(MWD/TO.U).

List of symbols used in this program:

IN	: Time step
NSYN	: Control index NSYN= -1,0,1
NXXX	: Punch index for region dependent densities for restart from a certain time step (NXXX= 0,1)
IR	: Region index
NLB	: Isotope index
DEN(L,IR)	: The atomic density of the isotopes for region IR
NDOW	: Group condensation control number
INSTOP	: Maximum time step
REACT	: Reactivity
L	: Isotope index
IX	: IN + 1
NRSTRT	: Time step number of density punch.

II- 13. SUBROUTINE GRUPVE

GRUPVE is called from the subroutine SYNOPS to perform the homogenization for the cell under consideration.

In this program, two different calculations are made for the option ISTO equal to and not equal to 1,.

If ISTO is equal to 1,(i.e, burn-up fuel cell calculation), the ratio of the average fuel to moderator flux (f^g) and homogenized density for the U-235 are computed as outlined below.

The ratio of the average fuel to moderator flux is given by,

$$f^g = \frac{\bar{\phi}_F^g}{\bar{\phi}_m^g} = \frac{\sum_{k=1}^{N} \phi_k^g V_k}{\sum_{k=N_{fuel}+1}^{N} \phi_k^g V_k} \quad \frac{V_m}{V_F} \qquad (2.29)$$

Where;

k , is the region index

$\bar{\phi}_F^g$; average fuel flux for group g

$\bar{\phi}_m^g$; average moderator flux for group g

V_m ; moderator volume

V_F ; fuel volume

This factor is calculated as follows;

First, the program calculates the total volume of the cell, total fuel volume, total flux times volume for the fuel regions and average flux for the fuel regions.

$$VGE = \sum_{IR=1}^{N20} VOL(IR)$$

$$VOLUM = \sum_{IR=1}^{IGC1} VOL(IR)$$

$$ANE2 = \sum_{IR=1}^{IGC1} \phi(IR,IE) \cdot VOL(IR)$$

$IE = 1, \ldots\ldots N26$

$$GAM(IE) = ANE2/VOLUM = \frac{\sum_{IR=1}^{IGC1} \phi(IE,IR) \cdot VOL(IR)}{\sum_{IR=1}^{IGC1} VOL(IR)}$$

Note that, the summations are over all fuel regions, for ISTO=1, the value of IGC1 was found as the number of regions in the fuel zone by the subroutine INPUTB (IGC1= MP(1))

Also, the program follows the same procedure for the moderator regions;

$$VOLUM = \sum_{IR=IGC2}^{N20} VOL(IR)$$

$$ANE2 = \sum_{IR=IGC2}^{N20} \phi(IE,IR) \cdot VOL(IR)$$

$$ANE2 = ANE2/VOLUM = \frac{\sum_{IR=IGC2}^{N20} \phi(IE,IR) \cdot VOL(IR)}{\sum_{IR=IGC2}^{N20} VOL(IR)}$$

Where IGC2= IGC1 + 1

Then the ratio of the average fuel to moderator flux is given by

$$ARGI(IE) = GAM(IE)/ANE2 = \frac{\sum_{IR=1}^{IGC1} \phi(IE,IR) \cdot VOL(IR) / \sum_{IR=1}^{IGC1} VOL(IR)}{\sum_{IR=IGC2}^{N20} \phi(IE,IR) \cdot VOL(IR) / \sum_{IR=IGC2}^{N20} VOL(IR)} \qquad (2.30)$$

In the final section of this routine, the homogenized density of 5 th isotope (U-235) over all fuel regions is computed,

$$ANE1 = \sum_{IR=1}^{IGC1} N(5,IR) \cdot VOL(IR)$$

$$DENLT = ANE1/VGE = \frac{\sum_{IR=1}^{IGC1} N(5,IR) \cdot VOL(IR)}{\sum_{IR=1}^{N20} VOL(IR)} \qquad (2.31)$$

Due to the low heterogenity it is a good approximation to use only two self-shielding factors for burnup supercell calculation,

$$f_F^g = \frac{\sigma^g}{1-\alpha\ (1-\sigma^g)}$$

$$f_M^g = \frac{1}{1-\alpha\ (1-\sigma^g)}$$

$$\alpha = \frac{V_F}{V_F + V_{NA}} \qquad (2.32)$$

for the nuclei mixed with the fuel and with the moderator, respectively. Within this coarse representation the canning material is added to the moderator. The symbol f^g denotes average fuel to moderator flux ratio as explained at the beginning of the subroutine. f^g is the carrier of the time dependence of the above equations. This will be described by the 5 th degree polynomial of a time dependent density of U-235.

$$f^g(t) = P_5\ (N(t)) \qquad (2.33)$$

The fuel cell burnup calculation (ISTO=1) prepares only the coefficients of the polynomial. Using these coefficients, the self shielding factors are calculated in burnup supercell calculation from two different formulas ; One for fuel (F), other for mod (M), afterwards, the self shielding factors (SSP), for the burnup supercell calculation are used in homogenization as explained in the following section. These results are written into file NUV(15) at the end of the section. When this routine is completed a return is made to the subroutine SYNOPS.

Generation of a cross section library for a fuel pin type surrounding an absorber in a supercell is accomplished by the following procedure. After a fuel pin transport calculation every cross section has to be homogenized by the individual self-shielding factor.

$$\Gamma_i^g = \frac{\sum_{k=1}^{N} N_k V_k \phi_k^g}{\sum_{k=1}^{N} N_k V_k} \cdot \frac{1}{\sum_{k=1}^{\tilde{N}} \phi_k^g V_k / \sum_{k=1}^{\tilde{N}} V_k} \qquad (2.34)$$

$$\Sigma_{x,i}^g = \Gamma_i^g \cdot \Sigma_{x,i}^g$$

i : isotope index

k : region "

x : index denoting the species of cross sections, differential ones included.

Using this factor homogenization will be made for the normal fuel cell calculation and control supercell calculation.

Maintaining the concept for the burn-up calculation of a supercell, however, would multipy the expenditure. Numerous tables of the fuel library depending on flux time had to be prepared before and to be read during the burnup calculation pushing the problem to unacceptable calculational times and storage demands.

In the second section of the GRUPVE, (ISTO ≠1), to perform the cell homogenization for all cases (i.e, except the burnup fuel cell calculation ISTO= 1) the following procedure is followed ;

The program calculates the volume of the homogenized cell, average flux for the cell, average density of all isotopes for the cell and self shielding factors for the cell,

$$VOLUM = \sum_{IR=IGC1}^{IGC2} VOL(IR)$$

where, IGC 1 : first region participating in homogenization

IGC 2 : last region participating in homogenization

$$ANE2 = \sum_{IR=IGC1}^{IGC2} \phi(IE,IR).\ VOL(IR).\ SSP$$

$$IE = 1,\ldots\ldots N26(45)$$

$$ABGI(IE) = ANE2/VOLUM = \frac{\sum_{IR=IGC1}^{IGC2} \phi(IE,IR).VOL(IR).SSP}{\sum_{IR=IGC1}^{IGC2} VOL(IR)} \qquad (2.35)$$

SSP_1 = DENN(IE,IR) the self shielding factor for all fuel regions specified in INPUT B

$$ANE1 = \sum_{IR=IGC1}^{IGC2} N(L,IR).\ VOL(IR)$$

$$DENH(L) = \frac{\sum_{IR=IGC1}^{IGC2} N(L,IR).\ VOL(IR)}{\sum_{IR=IGC1}^{IGC2} VOL(IR)} \qquad (2.36)$$

$$ZAE = \sum_{IR=IGC1}^{IGC2} N(L,IR).\ \phi(IE,IR).\ VOL(IR).SSP$$

$$GAM(IE,L) = ZAE/ANE1.\ ABGI(IE) = \frac{\sum_{IR=IGC2}^{IGC2} N(L,IR).\phi(IE,IR).VOL(IR).SSP}{\sum_{IR=IGC1}^{IGC2} N(L,IR)\,VOL(IR).\ \frac{\sum_{IR=IGC1}^{IGC2} \phi(IG,IR).VOL(IR).SSP}{\sum_{IR=IGC1}^{IGC2} VOL(IR)}}$$

Where,

$$SSP = DENN(IE,IR)/(1 + ALPHA.\ DEN(IE,IR) - 1) \qquad (2.37)$$

for the nuclei mixed with the fuel

If no supercell burnup calculation is present, SSP will be taken as 1 (SSP= 1 for all cases except the burnup supercell calculation) SSP, in the divisor of the above equation is equal to DENN(IE,IR) for burnup supercell calculations with, IE= 1 ,....N26.

All of the microscopic contants are multiplied by the self shielding factors as follows;

For fuel isotopes: L= 1,.....N FUEL, IE= 1,.....N26

SIGFI (IE,L)= SIGFI (IE,L).GAM (IE,L) $\left[\overline{\nu_f}(IE,L) . \overline{\Gamma}_L^{IE} \right]$

FISIG (IE,L)= FISIG (IE,L). GAM(IE,L) $\left[\nu\overline{\nu_f}(IE,L) . \overline{\Gamma}_L^{IE} \right]$

TOSIG (IE,L)= TOSIG (IE,L).GAM (IE,L) $\left[\overline{\nu_{tr}}(IE,L) . \overline{\Gamma}_L^{IE} \right]$

OUSIG (IE,1,L) = OUSIG (IE,1,L).GAM (IE,L) $\left[\overline{\nu_{sin}}(IE,L) . \overline{\Gamma}_L^{IE} \right]$

OUSIG (IE,2,L)= OUSIG (IE,2, L).GAM(IE,L) $\left[\overline{\nu_{sout}}(IE,L) \overline{\Gamma}_L^{IE} \right]$

For all isotopes: L= 1,.....N FUEL , IE= 1,.....N26

ABSIG (IE,L)= ABSIG(IE,L).GAM(IE,L) $\left[\overline{\nu_a}(IE,L) . \overline{\Gamma}_L^{IE} \right]$

For lumped absorber and nonburnable isotopes, moderator isotopes

TOSIG (IE,L)= TOSIG (IE,L).GAM (IE,L) $\left[\overline{\nu_{tr}}(IE,L) . \overline{\Gamma}_L^{IE} \right]$

$46 < L < NLB$

For lumped absorber and non burnable isotopes:

OUSIG(IE,1,L) = OUSIG(IE,1,L).GAM (IE,L) $\left[\overline{\nu_{sin}}(IE,L) . \overline{\Gamma}_L^{IE} \right]$

$46 < L < NLBM$ NLBM = NLB-NLM

OUSIG (IE,2,L) = OUSIG (IE,2,L).GAM(IE,L) $\left[\overline{\nu_{sout}}(IE,L) . \overline{\Gamma}_L^{IE} \right]$

All of the above microscopic cross sections homogenized by the self shielding factors are written to file NUX (22) if the case under consideration is pre-calculation of the control supercell calculation, (i.e, the option ISTO= 4).

For the option ISTO=4 , only homogenized library will be generated on logical file NUX. This, homogenized library generated by ISTO=4 will be used in control supercell calculation (ISTO=5).

However, the microscopic broad group cross sections for all isotopes in the cell under consideration are computed;

MIN = ICPU (IG) : no. of the highest small group
MAX = ICP(IG) : no. of the lowest small group
} within broad group NGR

ID = MIN.....MAX

$$ABGIJ = ABGI(IJ) = \bar{\phi}(IJ) = \frac{\sum_{IR=IGC1}^{IGC2} \phi(IJ,IR) \cdot VOL(IR) \cdot SSP}{\sum_{IR=IGC1}^{IGC2} VOL(IR)}$$

This is the average flux for each group in homogenized regions for the cell.

$$SNM = \sum_{IJ=MIN}^{MAX} ABGIJ = \sum_{IJ=MIN}^{MAX} \phi(IJ)$$

Total average flux for IG th broad group.

$$VFS = \sum_{IJ=MIN}^{MAX} FISIG(IJ,L) \cdot ABGIJ \left[\sum_{IJ=MIN}^{MAX} \nu\Sigma_f(IJ,L) \cdot \bar{\phi}(IJ)\right]$$

for $L \leq NFUEL$

$$VTS = \sum_{IJ=MIN}^{MAX} TOSIG(IJ,L) \cdot ABGIJ \left[\sum_{IJ=MIN}^{MAX} \Sigma_t(IJ,L) \cdot \bar{\phi}(IJ)\right]$$

for $L \leq NFUEL$ (fuel isotopes)

$$VTS = \sum_{IJ=MIN}^{MAX} TOSIG(IJ,L) \cdot ABGIJ \left[\sum_{IJ=MIN}^{MAX} \Sigma_t(IJ,L) \cdot \bar{\phi}(IJ)\right]$$

$46 < L < NLB$

Lumped absorbers, nonburnable isotopes, mod. isotopes.

$$VSIG1 = \sum_{IJ=MIN}^{MAX} SIGFI(IJ,L) \cdot ABGIJ \left[\sum_{IJ=MIN}^{MAX} \Sigma_f(IJ,L) \cdot \bar{\phi}(IJ)\right]$$

for $L \leq NFUEL$

$$VAS = \sum_{IJ=MIN}^{MAX} ABSIG(IJ,L) \cdot ABGIJ \left[\sum_{IJ=MIN}^{MAX} \Sigma_a(IJ,L) \cdot \bar{\phi}(IJ)\right]$$

for $L = 1,....,NLB$

Using the above quantities, the program calculates the broad group constants for all isotopes in the cell.

$$VFISI(IG,L) = VFS/SNM = \frac{\sum_{IJ:MIN}^{MAX} \nu\bar{\sigma}_f(IJ,L)\,\bar{\phi}(IJ)}{\sum_{IJ:MIN}^{MAX} \bar{\phi}(IJ)} \Big/ \left[\overline{\nu\Sigma_f}(IJ,L)\right]$$

$L \leq NFUEL$

$$VTOSI(IG,L) = VTS/SNM = \frac{\sum_{IJ:MIN}^{MAX} \bar{\sigma}_{tr}(IJ)\,\bar{\phi}(IJ)}{\sum_{IJ:MIN}^{MAX} \bar{\phi}(IJ)} \Big/ \left[\bar{\sigma}_{t}(IG,L)\right]$$

$L \leq NFUEL$

$$VTOSI(IG,L) = VTS/SNM = \frac{\sum_{IJ:MIN}^{MAX} \bar{\sigma}_{tr}(IJ,L)\,\bar{\phi}(IJ)}{\sum_{IJ:MIN}^{MAX} \bar{\phi}(IJ)} \Big/ \left[\bar{\sigma}_{tr}(IG,L)\right]$$

$46 < L < NLB$

(Lumped absorbers, nonburnable isotopes, mod.isotoper)

$$VABJI(IG,L) = VAS/SNM = \frac{\sum_{IJ:MIN}^{MAX} \bar{\sigma}_a(IJ,L)\,\bar{\phi}(IJ)}{\sum_{IJ:MIN}^{MAX} \bar{\phi}(IJ)} \left[\bar{F}_a(IG,L)\right]$$

$L = 1, \ldots\ldots NLB$ for all isotopes

$$VSIGFI(IG,L) = VSIG1/SNM = \frac{\sum_{IJ:MIN}^{MAX} \bar{\sigma}_f(IJ,L)\,\bar{\phi}(IJ)}{\sum_{IJ:MIN}^{MAX} \bar{\phi}(IJ)} \left[\bar{\sigma}_f(IG,L)\right]$$

$L \leq NFUEL$

$$VXN(IG,L) = VFISI(IG,L)/VSIGFI(IG,L) = \frac{\overline{\nu\sigma_f}(IG,L)}{\bar{\sigma}_f(IG,L)} \left[\bar{\nu}(IG,L)\right]$$

$L \leq NFUEL$

$$VOUSI(IG,2,L) = OUSIG(MAX,2,L)\cdot \bar{\phi}(MAX)/\sum_{IJ:MIN}^{MAX} \bar{\phi}(IJ);\ \left[\bar{\sigma}_{sout}(MAX,L)\right]$$

Also calculated are; The microscopic broad group in-scatter cross sections for fuel isotopes,

$$VOUSI(IG,1,L) = VTOSI(IG,L) - VABSI(IG,L) = VOUSI(IG,2,L)$$

The microscopic broad group in-scatter and out-scatter cross sections for lumped absorber and non-burnable isotopes.

$$VOUSI(IG,1,L) = OUSIG(MAX,2,L)\cdot \bar{\phi}(MAX)/\sum_{IJ:MIN}^{MAX} \phi(IJ)$$

$$VOUSI(IG,1,L) = VTOSI(IG,L) - VABSI(IG,L) - VOUSI(IG,2,L)$$

$$\bar{\sigma}_{sin}(IG,L) = \left[\bar{\sigma}_{tr}(IG,L) - \bar{\sigma}_a(IG,L) - \bar{\sigma}_{sout}(IG,L)\right]$$

Where $46 < L < NLBM$ indicating lumped absorber and non-burnable isotopes.

Finally, a return is made to the SYNOPS.

List of symbols used in this subroutine:

IR : region index

MI = 46 :

JFI = 31

VGE : total volume of the cell

VOLUM(IR) : the volume of the IR th region

IGC1 : first region participating in homogenization

IGC2 : last region participating in homogenization

IE : group index IE = 1,.....N26

RPHI(IE,IR): the flux for IE th group, IR th region

AVG1 (IE) : the homogenized flux over all regions for IE th group

DENN(IE,IR): the self shielding factor for IE th group and IR th region

DENH (L) : the homogenized density of the L th isotope in the cell

DENLT : the homogenized density of U-235

GAM (IE,L): the self shielding factor for IE th group and Lth isotopes

NFUEL : number of fuel isotopes

L >46 : represents the lumped absorber, nonburnable isotopes and moderator isotopes

MI <L ≤NLBM : represents the lumped absorber and non-burnable isotopes

NGR : number of broad groups.

MIN = ICPU (IG)
MAX = ICP (IG) } :represents the boundary of the broad group IG

SIGFI(IE,L): the microscopic fission cross section of the Lth fuel isotope for IEth group, times self shielding factor $\sigma_f \cdot \Gamma_L^\delta$

FISIG (IE,L): the microscopic $\nu \cdot \sigma_f$ for Lth fuel isotope and IE th group, times self shielding factor. $\left[\nu \sigma_f \cdot \Gamma_L^\delta \right]$

ABSIG (IE,L) : the microscopic absorption cross section for IE th group and L th isotope, times self shielding factor $[\Sigma_a \cdot \Gamma_L]$

TOSIG (IE,L) : the microscopic transport cross section for IE th group and L th isotope, times self shielding factor $[\Sigma_{tr} \Gamma_L]$

OUSIG (IE,1,L): the microscopic in-scatter cross section for IE th group and L th isotope, times self shielding factor

OUSIG (IE,2,L): the microscopic out-scatter cross section for IE th group and L th isotope, times self shielding factor

II- 14. SUBROUTINE GRUPVB

GRUPVB is referenced from the subroutine SYNOPS to calculate the macroscopic broad group cross sections and to write the homogenized microscopic broad group constants to IEREB file for use in the EREBUS code.

Firstly, the program reads data from a single card which is the control card for writing output that will be used for EREBUS input.

Then the microscopic and macroscopic broad group transfer cross sections of the moderator isotopes, broad group diffusion coefficients and the, macroscopic broad group constants for the cell are computed respectively.

Microscopic transfer cross section for moderator isotope is multiplied by the self shielding factor.

$$OUSIGM(IE,IJ,L) = GAMI \cdot OUSIGM(IE,IJ,L)\left[\Gamma_L^{IE} \cdot \sigma_s(IE \rightarrow IJ)\right]$$

Where, the factor GAMI $\left[\Gamma_L^{IE}\right]$ is calculated in the subroutine GRUPVE. for IE = 1,.....N26 IJ = 1,.....N26

If the option ISTO is equal to 4, these results are written into the file NUX (22) so as to use in the supercell calculation.

Microscopic broad group transfer cross section of the moderator isotope from broad group IGS to broad group IG is given by;

$$VOUSIM(IGS,IG,L) = \frac{\sum_{IE:MIN}^{MAX}\sum_{IJ:MIN}^{MAX} \sigma_s(IJ,IE,L)\,\bar{\phi}(IJ)}{\sum_{IJ:MIN}^{MAX} \bar{\phi}(IJ)} \; ; \left[\bar{\sigma}_s(IGS,IG,L)\right]$$

Where, $$\bar{\phi}(IJ) = \frac{\sum_{IR:IGC1}^{IGC2} \bar{\phi}(IJ,IR) \cdot VOL(IR) \cdot SSP}{\sum_{IR:IGC1}^{IGC2} VOL(IR)} \qquad IG = 1,\ldots NGR \quad IGS = 1,\ldots NGR$$

average flux over the homogenized regions for group IJ,

MII = ICPU (IG) lower limit of the broad group IG

MA = ICP (IG) upper limit of the broad group IG

MIN = ICPU(IGS) } lower and upper limits of the broad group IGS

MAX = ICP (IGS) }

Diffusion coefficients for broad groups are calculated in the same manner;

$$DIF1(IG) = \sum_{IE:MIN}^{MAX} DIF(IE).\ ANGI(IE) = \sum_{IE:MIN}^{MAX} DIF(IE).\ \bar{\phi}(IJ)$$

$$DIF1(IG) = DIF1(IG)/SNORMIG = \frac{\sum_{IE:MIN}^{MAX} DIF(IE).\ \bar{\phi}(IJ)}{\sum_{IJ:MIN}^{MAX} \bar{\phi}(IJ).} \qquad (2.38)$$

Where, DIF(IE) is the diffusion coefficient for the microgroup IE and is computed in the subroutine DEPRIN.

Multiplying the microscopic broad group constants which were generated in the subroutine GRUPVE by the homogenized densities for the cell, macroscopic broad group constants are obtained as follows;

The homogenized density of every isotope in the cell is given by,

$$\bar{N}(L) = DENH(L) = \frac{\sum_{IR:IGC1}^{IGC2} N(L,IR).\ VOL(IR)}{\sum_{IR:IGC1}^{IGC2} VOL(IR)}$$

The microscopic broad group cross sections created in GRUPVE for each isotope is given by,

$$\overline{\sigma_X}(IG,L) = \frac{\sum_{IJ:MIN}^{MAX} \sigma_X(IJ.L).\ \bar{\phi}(IJ)}{\sum_{IJ:MIN}^{MAX} \bar{\phi}(IJ)} \qquad (2.39)$$

Where, IG : Broad group index : IG = 1, ...NGR

x : Index denoting the type of cross sections;

$\bar{\sigma}_f(IG,L)$, $\bar{\sigma}_t(IG,L)$, $\bar{\sigma}_a(IG,L)$, $\bar{\sigma}_{sin}(IG,L)$, $\bar{\sigma}_{sout}(IG,L)$

Then, the macroscopic broad group cross sections are computed.

MACROSCOPIC BROAD GROUP FISSION CROSS SECTION; $\Sigma_f(IG)$;

$$VSIG(IG) = \sum_{L=1}^{NFUEL} DENH(L) \cdot \bar{\sigma}_f(IG,L)$$

MACROSCOPIC BROAD GROUP TRANSPORT CROSS SECTION : $\Sigma_t(IG)$:

$$TOS(IG) = \sum_{L=1}^{NFUEL} DENH(L) \cdot \bar{\sigma}_t(IG,L) + \sum_{L=47}^{NLB} DENH(L) \cdot \bar{\sigma}_t(IG,L)$$

MACROSCOPIC BROAD GROUP ABSORPTION CROSS SECTION; $\Sigma_a(IG)$;

$$ABS(IG) = \sum_{L=1}^{NLB} DENH(L) \cdot \bar{\sigma}_a(IG,L)$$

MACROSCOPIC BROAD GROUP IN SCATTERING CROSS SECTION: $\Sigma_{sin}(IG)$;

$$OUMM(IG,L) = \sum_{L=1}^{NFUEL} DENH(L) \cdot \bar{\sigma}_{sin}(IG,L) + \sum_{L=47}^{NLBM} DENH(L) \cdot \bar{\sigma}_{sin}(IG,L)$$

Where, NLBM = NLB- NLM : number of isotopes except the moderator isotopes.

MACROSCOPIC BROAD GROUP OUT SCATTERING CROSS SECTION; $\Sigma_{sout}(IG)$;

$$OUMM(IG,2) = \sum_{L=1}^{NFUEL} DENH(L) \cdot \bar{\sigma}_{sout}(IG,L) + \sum_{L=47}^{NLBM} DENH(L) \cdot \bar{\sigma}_{sout}(IG,L)$$

MACROSCOPIC BROAD GROUP TRANSFER CROSS SECTION, $\Sigma_s(IG,IGS)$;

$$OUM(IG,IGS) = \sum_{ILM=1}^{NLM} DENH(ILM) \cdot \bar{\sigma}_s(IG,IGS,L)$$

NLM: number of moderator isotopes.

MACROSCOPIC BROAD GROUP REMOVAL CROSS SECTION, $\sum r(IG)$;

SREM (IG) = OUM (IG + 1, IG)

On the other hand; the microscopic broad group cross sections for all isotopes are written into the file IEREB for use in EREBUS code. Recall that this cross-section library is generated by the subroutine GRUPVE.

However, the program prints in the output, the microscopic and macroscopic broad group cross sections.

If the case under consideration is a supercell calculation the program calculates the effective cross sections and prints them in the output. The calculation is given by,

EFGQ1 = DENH(L) . VASSI(IG,L) $\left[\bar{N}(L).\overline{\sigma_a}(IG,L)\right]$ (2.40)

Where L = 41,....49 lumped absorber isotopes.

DENH(L) : the density of the lumped absorber isotopes

For the same case, the shielding factors of the burnable poisons and cell fluxes are printed.

When this program is completed, a return is made back to SYNOPS.

List of symbols used in this program:

NLM : number of moderator isotopes

IE : group index IE = 1,.....N26

IJ : group index IJ = 1,.....N26

IG,IGS : broad group index. IG = 1,...NGR, IGS = 1,...NGR

DIF1(IG) : diffusion cross section for the broad groupe IG

VSIG (IG) : macroscopic broad group fission cross section for the cell

FIS (IG) : macroscopic broad group nu.fission cross section for the cell

TOS (IG) : macroscopic broad group transport cross section for the cell

ABS (IG) : macroscopic broad group absorption cross section for the cell

OUMM(IG,1): macroscopic broad group in-scatter cross section for the cell

OUMM(IG,2): macroscopic broad group out-scatter cross section for the cell

OUM (IG,IGS):macroscopic broad group transfer cross section from IG th broad group to broad group IGS

SREM (IG) : macroscopic removal cross section for broad group IG

GAMI = GAM(IE,L) : the self shielding factor for IE th micro group and L th isotope.

NGR : number of broad groups

NLB : number of isotopes

NFUEL : number of fuel isotopes

DENH (L) : represents the homogenized density of the isotope L.

VOLUM : total volume of the homogenized region.

NPUNC : control index for writing the input to EREBUS

NLBM = NLB - NLM : number of isotopes except the moderator isotopes

MI = 46

DFI = 31

II- 15. SUBROUTINES ORDO AND COBES

If burnup fuel cell calculation (ISTO = 1) is to be considered, the subroutine COBES is called from the MAIN program only to calculate a set of polynomial coefficients which correlate the self shielding factors in each energy group with the U-235 concentration.

$$f^{g}(t) = P_{g}\ (\ N(t)\ )$$

For this purpose the program COBES is called from the MAIN for U-235 isotope only.

If no burn-up fuel cell calculation (ISTO $\neq$ 1) is to be considered, then both COBES and ORDO are referenced from the MAIN together. The subroutine ORDO reads the homogenized density and self shielding factors for all isotopes from file NUZ and writes them to file NUV for use in COBES.

The subroutine COBES is called from the MAIN for each isotope within the cell seperately to calculate the burn-up dependence of self shielding factors using the least squares orthogonel polynomial technique.

The appropriate function for the least squares fit is a sum of orthogonal polynomials* which has the form,

$$y(x) = a + b\ (x-\beta) + c\ (x-\gamma_1)(x-\gamma_2) + d\ (x-S_1)(x-S_2)(x-S_3) + \ldots . \quad (2.41)$$

Where, y: the dependent variable

a, b, c, d : coefficients

x: the independent variable

---

*) Any polynomial can be written as a sum of orthogonal polynomials

$y = a + \sum_{j=1}^{n} \left[b_j\ X_j(x_i) \right]$ with the orthogonal property that,

$$\sum_{i} \left[x_j(x_i)\ X_k(x_i) \right] = 0 \quad \text{for } j \neq k$$

χ^2 is defined as,

$$\chi^2 \equiv \sum_i \left(\frac{\Delta y_i}{\sigma_i}\right)^2 = \sum_i \left\{ \frac{1}{\sigma_i^2} \left[y_i - y(x_i) \right]^2 \right\} \quad : \qquad (2.42)$$

Where,

σ_i: the standard deviation

$$\sigma_i = \sqrt{\left(\frac{1}{N} \sum_i x_i^2\right) - \bar{x}^2}$$

N : number of observations

χ^2 must be minimized to determine the coefficients a, b, c, d, etc. with the further criterion that the addition of higher-order terms to the polynomial will not effect the evaluation of lower-order terms. This criterion will be used to determine the parameters β, γ_1, γ_2, etc.

Setting the derivatives of χ^2 with respect to each of the coefficients equal to 0 yields n + 1 simultaneous equations.

$$\sum_i y_i = Na + b\sum(x_i-\beta) + c\sum(x_i-\gamma_1)(x_i-\gamma_2) + d\sum(x_i-\delta_1)(x_i-\delta_2)(x_i-\delta_3) + \ldots. \qquad (2.43)$$

$$\sum_i x_i y_i = a\sum x_i + b\sum x_i(x_i-\beta) + c\sum x_i(x_i-\gamma_1)(x_i-\gamma_2) + d\sum x_i(x_i-\delta_1)(x_i-\delta_2)(x_i-\delta_3) + \ldots. \qquad (2.44)$$

$$\sum_i x_i^2 y_i = a\sum x_i^2 + b\sum x_i^2(x_i-\beta) + c\sum x_i^2(x_i-\gamma_1)(x_i-\gamma_2) + d\sum x_i^2(x_i-\delta_1)(x_i-\delta_2)(x_i-\delta_3) + \ldots. \qquad (2.45)$$

$$\sum_i x_i^3 y_i = a\sum x_i^3 + b\sum x_i^3(x_i-\beta) + c\sum x_i^3(x_i-\gamma_1)(x_i-\gamma_2) + d\sum x_i^3(x_i-\delta_1)(x_i-\delta_2)(x_i-\delta_3) + \ldots. \qquad (2.46)$$

We have omitted a factor of σ_i^2 in the denominator of each term for clarity.

If one is restricted to a 0th degree polynomial, there is only one coefficient a ; all of the other coefficients are set equal to 0 by definition. The coefficient a, therefore has the form

$$a = \frac{1}{N} \sum y_i = \bar{y} \qquad (2.47)$$

If on the other hand a first-degree polynomial is under consideration, the coefficient b is not 0. In order to have independence of coefficients, however, this term must still be 0. Hence the conclusion that the sum in this term is 0.

$$\sum (x_i - \beta) = 0$$

leads to a value for β

$$\beta = 1/N . \sum x_i = \bar{x} \qquad (2.48)$$

Similarly, if we consider a quadratic function, the third term of equation (2.43) must be 0 even when the coefficient C is not 0. This constraint leads to a quadratic equation in γ_1 and γ_2. We have the additional constraint, however that the coefficient b must be specified by equation (2.43) and (2.44), that is equation (2.44) must be determined after a is determined (2.43). Thus, the third term in equation (.244) must also be 0 regardless of the value of the coefficient c .

$$\sum_i (x_i - \gamma_1)(x_i - \gamma_2) = 0$$

$$\sum_i x_i (x_i - \gamma_1)(x_i - \gamma_2) = 0 \qquad (.2.49)$$

Similarly

$$\sum_i x_i (x_i - \delta_1)(x_i - \delta_2)(x_i - \delta_3) = 0$$

$$\sum x_i(x_i-\delta_1)(x_i-\delta_2)(x_i-\delta_3)=0$$

$$\sum x_i^2(x_i-\delta_1)(x_i-\delta_2)(x_i-\delta_3)=0$$

The extrapolation to higher-order parameters is straight forward.

Once the parameters β, γ, δ, etc. are determined by the constraints described above, the coefficients a, b, c, etc. can be determined from the resulting (n + 1) simultaneous equations. The value for the first coefficient a is specified completely by minimizing χ^2 with respect to a in equation (2.43). The value of the second coefficient b is determined by minimizing χ^2 with respect to both a and b in equations (2.43 and 2.44). Substituting the value of a into equation (2.44) yields a result for b directly. Similarly, the value for c can be determined from equation (2.45), after substituting the values of a and b determined from equations (2.43 and 2.44). Each succeeding equation yields the value for the next higher-order coefficient.

$$a=\frac{1}{N}\sum y_i=\bar{y}$$

$$b=\sum y_i(x_i-\beta)/\sum(x_i-\beta)^2$$

$$c=\frac{\sum\left[y_i(x_i-\gamma_1)(x_i-\gamma_2)\right]}{\sum\left[(x_i-\gamma_1)(x_i-\gamma_2)\right]} \qquad (.250)$$

$$d=\frac{\sum\left[y_i(x_i-\delta_1)(x_i-\delta_2)(x_i-\delta_3)\right]}{\sum\left[(x_i-\delta_1)(x_i-\delta_2)(x_i-\delta_3)\right]^2}$$

and so on,

CHAPTER III

NUMERICAL APPLICATION

III- 1. INPUT DATA PREPARATION

For one-dimensionel burnup and spectrum calculation the following fuel cell model is used. The fuel cell is divided into four zones in order to achive a straight forward one dimensional treatment of the whole micro structure.

ZONE 1: Consists of fuel (UO_2-ThO_2) and gas gap, radius of the zone is equal to the radius of the cladding inner surface. Total fuel density :

$$\rho_{fuel} = f_{ThO_2} \cdot \rho_{ThO_2} + (1 - f_{ThO_2}) \cdot \rho_{UO_2} \qquad (3.1)$$

Where,

f_{ThO_2} : fraction of ThO_2 in total fuel

ρ_{ThO_2} : density of ThO_2 (gr/cm^3)

ρ_{UO_2} : density of UO_2 (gr/cm^3)

This total density is reduced by the factor of $(1-0.015)\frac{r_p^2}{r_1^2}$ to include also the gas gap and dishing effect into the fuel region, rp being the radius of the fuel pin, r_{c1} being the cladding inner radius, both given in cm.

New densities of ThO_2 and UO_2 :

$$\rho_{ThO_2} = \rho_{fuel} \cdot f_{ThO_2}$$

$$\rho_{UO_2} = \rho_{fuel} \cdot (1 - f_{ThO_2})$$

The fraction of U and Th in UO_2 and ThO_2 respectively :

$$f_x = \frac{E\ MA + (1-E)\ MB}{E\ MA + (1-E)\ MB + MO} \qquad (3.2)$$

Whose,

E : fissile enrichment in the Uranium material (%)

MA : atomic mass of the fissile element

MB : atomic mass of the non-fissile element

MO : atomic mass of the oxygen

and, subscript x = either U or Th

The density of each element:

$$\rho_{Th} = f_{Th} \cdot \rho_{ThO_2}$$

$$\rho_{O_2}^{Th} = \rho_{ThO_2} - \rho_{Th}$$

$$\rho_U = f_U \cdot \rho_{UO_2}$$

$$\rho_{235} = E \cdot \rho_U$$

$$\rho_{238} = (1-E) \cdot \rho_U$$

$$\rho_{O_2}^{U} = \rho_{UO_2} - \rho_U$$

$$\rho_{Total}^{O_2} = \rho_{O_2}^{Th} + \rho_{O_2}^{U}$$

After having found the densities the number densities are found by;

$$N = \frac{\rho \cdot NAV}{M} \qquad (3.3)$$

Where,

M : atomic mass of the element

NAV : Avagadro number

NAV : $0{,}602252 \cdot 10^{24}$ Nuclei/mol = 0,602252 szilard

ZONE 2: Consists of cladding (Zr-4). The radius of this zone is equal to cladding outside radius.

radius of zone 2 = 0.475 cm.

composition of zircaloy 4

The number densities of the elements can be calculated from the equation:

$$N = \frac{\rho_{Zr-4} \cdot NAV}{M_{Zr-4}} f_{an} \qquad (3.4)$$

Where,

$\rho_{Zr-4} = 6{,}55$ gr/cm^3

$M_{Zr-4} = 91{,}3375$ (atomic weight of zircaloy-4)

For the calculation of number densities of other zones, see (1)

III- 2. DISCUSSION OF THE NUMERICAL RESULTS

In this study, for cells with different mixtures of UO_2 - ThO_2 and different enrichment levels of the U-235 component normal cell calculation, burnup fuel cell and burnup super cell calculations have been made using the computer code GELS.

As a result, microscopic and macroscopic broad group constants, weights of fuel elements, reactivity, fast and thermal group fluxes, homogenized densities were obtained for the types of cells examined. The main objective was to determine the way in which the number densities of important isotopes and k_∞ change with time steps for both fuel and super cells during one year burnup period.

Table (III- 2.1) shows the results as an example, for ThO_2- UO_2 mixture rations of 30% - 70% and 40% - 60% by weight. In both mixtures k_∞ increases with increasing enrichment while the normalized cell fluxes decrease as expected. This is because richer fuel would sustain a lower flux level for the same power production. In case of 30% UO_2 - 70% ThO_2 mixture 18% W/U enrichment level for the UO_2 component results in $k_\infty = 1$. This type of a cell being relatively rich in Th, would have a high conversion ratio and a high rate of U-233 production but obviously would have insufficient excess reactivity for control purposes and from the point of view of core life time.

Table (III-2.2) shows the results for 70% UO_2 - 30% ThO_2 mixture. This type of a cell, being relatively poor in Th, would have a low conversion ratio therefore the U-233 production is decreased, but would have significiant excess reactivity for core life time.

Tables (III- 2.5,6,7) show the change in the atomic density of xenon-135 nuclide depending on the time steps during one year burnup period for various mixtures of UO_2 - ThO_2 and the enrichment levels of the UO_2 component.

It is seen that, the atomic density of xenon-135 nuclide increases with increasing amounts of UO_2 and enrichment levels of the UO_2 component as expected.

Also, as seen in tables (III- 2,5,6,7) and figures (5,6,7) the atomic density of xenon-135 decreases with increasing time steps. The concentration of xenon-135 is reaches its maximum value at the end of first month and then begins to decrease with time steps. Because xenon-135 has a very large absorption cross section for thermal neutrons, low xenon-135 concentration is desirable for control purposes especially for start up and shutdown.

Figure (8) shows the U-233 production in grams for one year burnup period versus the enrichment levels for different mixtures of UO_2- ThO_2. It is seen that the U-233 production decreases by increasing amounts of both UO_2 and the enrichment of the UO_2 component as expected.

In the extreme case of pure UO_2 cells we have k_∞ exceeding 1 by more than enough margins for control and sufficiently long core life time purposes. However such cells wouldn't be of any use in Th conversion but would be the main source of energy production in the core.

Clearly a core that is to produce energy on a commercial scale while converting significant amounts of Th would have to be made up of a mixture of the different types of cells to be employed in the finite reactor geometry. Factors that would guide ones efforts would be ; relatively flat power production, desirable levels of Th conversion, sufficiently long reactor core life time and ease of control during start up and shut down. In a desingn process different core configurations would be investigated with the data generated in this study and the configurations that optimize the above factors would be chosen as the design basis.

Hence further work to be suggested along the lines of this study would be to use a reactor analysis code such as EREBUS utilizing the data generated by GELS here to design a commercial PWR core converting significiant amounts Th without much sacrifice on other performance criteria.

Table (III- 2.1) The results for fuel cell calculation

FUEL CELL CALCULATION $f_{UO_2} = 0.30$, $f_{ThO_2} = 0.70$			
Enrichment % U-235	keff	cell flux	
		1st group	2nd group
10	0,83193	$3,66915.10^{14}$	$5,95353.10^{13}$
12	0,89006	$3,41644.10^{14}$	$5,07024.10^{13}$
14	0,93721	$3,23264.10^{14}$	$4,26692.10^{13}$
16	0,97629	$3,09214.10^{14}$	$3,93519.10^{13}$
18	1,0093	$2,98068.10^{14}$	$3,54571.10^{13}$
20	1,0375	$2,88965.10^{14}$	$3,22854.10^{13}$

FUEL CELL CALCULATION $f_{UO_2} = 0.40$, $f_{ThO_2} = 0.60$			
Enrichment % U-235	keff	cell flux	
		1st group	2nd group
10	0,93808	$3,23753.10^{14}$	$4,58225.10^{13}$
12	0,99082	$3,05038.10^{14}$	$3,90421.10^{13}$
14	1,0328	$2,91281.10^{14}$	$3,40751.10^{13}$
16	1,0671	$2,80643.10^{14}$	$3,02594.10^{13}$
18	1,0957	$2,72097.10^{14}$	$2,7231.10^{13}$
20	1,12014	$2,65022.10^{14}$	$2,47404.10^{13}$

Table (III-2.2) The results for fuel cell calculation

FUEL CELL CALCULATION f_{UO_2}: 0.50, f_{ThO_2}:0.50			
Enrichment % U-235	keff	cell flux	
		group 1	group 2
10	1,0102	$2,98546.10^{14}$	$3,74353.10^{13}$
12	1,0579	$2,83432.10^{14}$	$3,18682.10^{13}$
14	1,0955	$2,72194.10^{14}$	$2,77769.10^{13}$
16	1,1260	$2,63376.10^{14}$	$2,46174.10^{13}$
18	1,1515	$2,56188.10^{14}$	$2,20948.10^{13}$
20	1,1733	$2,50147.10^{14}$	$2,00269.10^{13}$

FUEL CELL CALCULATION f_{UO_2}:0.70, f_{ThO_2}:0.30			
Enrichment % U-235	keff	cell flux	
		group 1	group 2
10	1,1085	$2,68218.10^{14}$	$2,73620.10^{13}$
12	1,1481	$2,56991.10^{14}$	$2,32078.10^{13}$
14	1,1794	$2,48354.10^{14}$	$2,01277.10^{13}$
16	1,2051	$2,41360.10^{14}$	$1,77400.10^{13}$
18	1,2268	$2,35480,10^{14}$	$1,58271.10^{13}$
20	1,2456	$2,30391.10^{14}$	$1,42551.10^{13}$

Table (III- 2.3) The result for fuel cell calculation

FUEL CELL CALCULATION f_{UO_2} :1,00 , f_{ThO_2} : 0			
Enrichment % U-235	keff	cell flux	
		group 1	group 2
10	1,1999	$2,42953.10^{14}$	$1,94048.10^{13}$
12	1,2326	$2,34155.10^{14}$	$1,63224.10^{13}$
14	1,2592	$2,27030.10^{14}$	$1,40242.10^{13}$
16	1,2814	$2,2145.10^{14}$	$1,19274.10^{13}$
18	1,3016	$2,15756.10^{14}$	$1,08176.10^{13}$
20	1,3192	$2,11014.10^{14}$	$9,63362.10^{12}$

Table (III-2.4) U-233 Production for one year in gr. for the cell

Enrichment % U-235	30% UO_2-70% ThO_2	50% UO_2-50% ThO_2	70% UO_2-30% ThO_2
10	$2,4823.10^{-2}$	$1,3341.10^{-2}$	$6,807.10^{-3}$
12	$2,2859.10^{-2}$	$1,2247.10^{-2}$	$6,2673.10^{-3}$
16	$1,9931.10^{-2}$	$1,0719.10^{-2}$	$5,5135.10^{-3}$
20	$1,7917.10^{-2}$	$9,6876.10^{-3}$	$4,9982.10^{-3}$

Table (III-2.5) Atomic density of xenon-135 depending on time steps

Time Steps	30% UO_2 - 70% ThO_2			
	E = 10%	E = 12	E = 16	E = 20
1	$7,16809.10^{-9}$	$8,41093.10^{-9}$	$1,07068.10^{-8}$	$1,27514.10^{-8}$
2	$7,18322.10^{-9}$	$8,43842.10^{-9}$	$1,07610.10^{-8}$	$1,28276.10^{-8}$
3	$7,14852.10^{-9}$	$8,39134.10^{-9}$	$1,0680.10^{-8}$	$1,27773.10^{-8}$
4	$7,12802.10^{-9}$	$8,34865.10^{-9}$	$6,91974.10^{-9}$	$1,27089.10^{-8}$
5	$7,11862.10^{-9}$	$8,3151.10^{-9}$	$7,03428.10^{-9}$	$1,26398.10^{-8}$
6	$7,11111.10^{-9}$	$8,28343.10^{-9}$	$7,12599.10^{-9}$	$1,25712.10^{-8}$
7	$7,10173.10^{-9}$	$8,25024.10^{-9}$	$7,19994.10^{-9}$	$1,25013.10^{-8}$
8	$7,08906.10^{-9}$	$8,21438.10^{-9}$	$7,25941.10^{-9}$	$1,24295.10^{-8}$
9	$7,07291.10^{-9}$	$8.17564.10^{-9}$	$7,30706.10^{-9}$	$1,23554.10^{-8}$
10	$7,05334.10^{-9}$	$8,13397.10^{-9}$	$7,34458.10^{-9}$	$1,22792.10^{-8}$
11	$7,03079.10^{-9}$	$8,08965.10^{-9}$	$7,37370.10^{-9}$	$1,22007.10^{-8}$
12	$7,00551.10^{-9}$	$8,04307.10^{-9}$	$7,39539.10^{-9}$	$1,21202.10^{-8}$

Table (III- 2.6) Atomic density of xenon-135 depending on time steps

Time Steps	70% UO_2 - 30% ThO_2			
	E = 10	E = 12	E = 16	E= 20
1	$1,43097.10^{-8}$	$1,61796.10^{-8}$	$1,91274.10^{-8}$	$2,12465.10^{-8}$
2	$1,44518.10^{-8}$	$1,63259.10^{-8}$	$1,92633.10^{-8}$	$2,13608.10^{-8}$
3	$1,44698.10^{-8}$	$1,63409.10^{-8}$	$1,92753.10^{-8}$	$2,13709.10^{-8}$
4	$1,44576.10^{-8}$	$1,63281.10^{-8}$	$1,92693.10^{-8}$	$2,13718.10^{-8}$
5	$1,44344.10^{-8}$	$1,63022.10^{-8}$	$1,92489.10^{-8}$	$2,13603.10^{-8}$
6	$1,44061.10^{-8}$	$1,62708.10^{-8}$	$1,92214.10^{-8}$	$2,13419.10^{-8}$
7	$1,43738.10^{-8}$	$1,62359.10^{-8}$	$1,91903.10^{-8}$	$2,13192.10^{-8}$
8	$1,43337.10^{-8}$	$1,61978.10^{-8}$	$1,91564.10^{-8}$	$2,12938.10^{-8}$
9	$1,42980.10^{-8}$	$1,61570.10^{-8}$	$1,91205.10^{-8}$	$2,12662.10^{-8}$
10	$1,42550.10^{-8}$	$1,61133.10^{-8}$	$1,90827.10^{-8}$	$2,12371.10^{-8}$
11	$1,42084.10^{-8}$	$1,60671.10^{-8}$	$1,90433.10^{-8}$	$2,12067.10^{-8}$
12	$1,41590.10^{-8}$	$1,60185.10^{-8}$	$1,90021.10^{-8}$	$2,11751.10^{-8}$

Table (III, 2.7) Atomic density of xenon-135 depending on time steps

Time Steps	50% UO_2 - 50% ThO_2			
	E = 10	E = 12	E = 16	E = 20
1	$1,10592.10^{-8}$	$1,27469.10^{-8}$	$1,56289.10^{-8}$	$1,79322.10^{-8}$
2	$1,11492.10^{-8}$	$1,28524.10^{-8}$	$1,58513.10^{-8}$	$1,80557.10^{-8}$
3	$1,11370.10^{-8}$	$1,28378.10^{-8}$	$1,57398.10^{-8}$	$1,80488.10^{-8}$
4	$1,11050.10^{-8}$	$1,27989.10^{-8}$	$1,57030.10^{-8}$	$1,80219.10^{-8}$
5	$1,10708.10^{-8}$	$1,27552.10^{-8}$	$1,56559.10^{-8}$	$1,79816.10^{-8}$
6	$1,10340.10^{-8}$	$1,27090.10^{-8}$	$1,56051.10^{-8}$	$1,79360.10^{-8}$
7	$1,10140.10^{-8}$	$1,26597.10^{-8}$	$1,55518.10^{-8}$	$1,78872.10^{-8}$
8	$1,09780.10^{-8}$	$1,26071.10^{-8}$	$1,54960.10^{-8}$	$1,78363.10^{-8}$
9	$1,09460.10^{-8}$	$1,25511.10^{-8}$	$1,54378.10^{-8}$	$1,77832.10^{-8}$
10	$1,0914.10^{-8}$	$1,24921.10^{-8}$	$1,53770.10^{-8}$	$1,77284.10^{-8}$
11	$1,0824.10^{-8}$	$1,24299.10^{-8}$	$1,53140.10^{-8}$	$1,76718.10^{-8}$
12	$1,0745.10^{-8}$	$1,23649.10^{-8}$	$1,52487.10^{-8}$	$1,76134.10^{-8}$

Table(III-2-8):The weights of fuel isctopesfor 30%UO_2-70%THO_2 fuel mixture and 10%U-235 enrichment level

a)

TIME DAYS	U234 GRAMS	U235 GRAMS	U236 GRAMS	U237 GRAMS	NP237 GRAMS	BURNUP MWD/TO-U
.00	0.0000	1.4045-01	0.0000	0.0000	0.0000	0.0000
4.17	7.4049-07	1.3952-01	1.9022-04	1.0119-07	1.5711-08	1.5335+02
0.00	3.4201-05	1.3386-01	1.3497-03	2.7403-06	3.3631-06	1.1042+03
0.00	1.1989-04	1.2749-01	2.6457-03	6.6476-06	1.7560-05	2.2083+03
0.00	2.4047-04	1.2138-01	3.8773-03	1.0493-05	4.3116-05	3.3125+03
0.00	3.8703-04	1.1554-01	5.0422-03	1.4182-05	7.9285-05	4.4166+03
0.00	5.5446-04	1.0998-01	6.1415-03	1.7702-05	1.2523-04	5.5208+03
0.00	7.3958-04	1.0468-01	7.1777-03	2.1056-05	1.8012-04	6.6249+03
0.00	9.4023-04	9.9637-02	8.1538-03	2.4251-05	2.4318-04	7.7291+03
0.00	1.1547-03	9.4829-02	9.0725-03	2.7296-05	3.1366-04	8.8332+03
0.00	1.3817-03	9.0247-02	9.9368-03	3.0195-05	3.9088-04	9.9374+03
0.00	1.6200-03	8.5880-02	1.0749-02	3.2956-05	4.7415-04	1.1042+04
0.00	1.8685-03	8.1718-02	1.1513-02	3.5584-05	5.6287-04	1.2146+04

b)

TIME DAYS	K-EFFEKT.	CONVERS. RATIO	TH232 GRAMS	PA233 GRAMS	U233 GRAMS	BURNUP MWD/TO-U
.00	8.3193-01	0.0000	3.3774+00	0.0000	0.0000	0.0000
4.17	8.0859-01	1.6334-01	3.3770+00	4.3874-04	2.3477-05	1.5335+02
0.00	7.9468-01	4.3170-01	3.3740+00	2.3436-03	9.8155-04	1.1042+03
0.00	7.8660-01	5.5739-01	3.3705+00	3.4565-03	3.1172-03	2.2083+03
0.00	7.8110-01	6.1861-01	3.3669+00	3.9873-03	5.7016-03	3.3125+03
0.00	7.7638-01	6.5115-01	3.3633+00	4.2438-03	8.4082-03	4.4166+03
0.00	7.7209-01	6.7047-01	3.3597+00	4.3715-03	1.1089-02	5.5208+03
0.00	7.6800-01	6.8363-01	3.3561+00	4.4391-03	1.3679-02	6.6249+03
0.00	7.6401-01	6.9309-01	3.3525+00	4.4792-03	1.6152-02	7.7291+03
0.00	7.6007-01	7.0276-01	3.3489+00	4.5068-03	1.8502-02	8.8332+03
0.00	7.5617-01	7.1099-01	3.3452+00	4.5290-03	2.0727-02	9.9374+03
0.00	7.5229-01	7.1884-01	3.3415+00	4.5491-03	2.2833-02	1.1042+04
0.00	7.4845-01	7.2649-01	3.3379+00	4.5687-03	2.4823-02	1.2146+04

c)

KFO ENRICHMENT 2.6%

NE TEP	TIME DAYS	POIS1 GRAMS	POIS2 GRAMS	POIS3 GRAMS	U238 GRAMS	NP239 GRAMS	BUNUP MWD/TO-U
0	.00	0.0000	0.0000	0.0000	1.2641+00	0.0000	0.0000
1	4.17	0.0000	0.0000	0.0000	1.2639+00	8.6275-05	1.5335+02
2	30.00	0.0000	0.0000	0.0000	1.2628+00	1.2405-04	1.1042+03
3	60.00	0.0000	0.0000	0.0000	1.2616+00	1.2630-04	2.2083+03
4	90.00	0.0000	0.0000	0.0000	1.2603+00	1.2764-04	3.3125+03
5	120.00	0.0000	0.0000	0.0000	1.2590+00	1.2856-04	4.4166+03
6	150.00	0.0000	0.0000	0.0000	1.2577+00	1.2929-04	5.5208+03
7	180.00	0.0000	0.0000	0.0000	1.2564+00	1.2995-04	6.6249+03
8	210.00	0.0000	0.0000	0.0000	1.2551+00	1.3057-04	7.7291+03
9	240.00	0.0000	0.0000	0.0000	1.2538+00	1.3118-04	8.8332+03
0	270.00	0.0000	0.0000	0.0000	1.2525+00	1.3179-04	9.9374+03
1	300.00	0.0000	0.0000	0.0000	1.2511+00	1.3239-04	1.1042+04
2	330.00	0.0000	0.0000	0.0000	1.2498+00	1.3300-04	1.2146+04

d)

KFO ENRICHMENT 2.6%

NE TEP	TIME DAYS	PU239 GRAMS	PU240 GRAMS	PU241 GRAMS	PU242 GRAMS	AM243 GRAMS	BURNUP MWD/TO-U
0	.00	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000
1	4.17	6.3758-05	3.0485-07	0.0000	0.0000	0.0000	1.5335+02
2	30.00	9.1398-04	2.4617-05	1.9412-06	1.8860-08	0.0000	1.1042+03
3	60.00	1.8197-03	9.4338-05	1.5305-05	3.1742-07	3.9265-09	2.2083+03
4	90.00	2.6030-03	1.9297-04	4.6946-05	1.5177-06	2.9097-08	3.3125+03
5	120.00	3.2792-03	3.0842-04	9.8766-05	4.3931-06	1.1437-07	4.4166+03
6	150.00	3.8630-03	4.3259-04	1.6969-04	9.6989-06	3.2202-07	5.5208+03
7	180.00	4.3676-03	5.6015-04	2.5719-04	1.8086-05	7.3597-07	6.6249+03
8	210.00	4.8043-03	6.8760-04	3.5816-04	3.0071-05	1.4547-06	7.7291+03
9	240.00	5.1825-03	8.1268-04	4.6946-04	4.6031-05	2.5893-06	8.8332+03
0	270.00	5.5103-03	9.3391-04	5.8817-04	6.6211-05	4.2576-06	9.9374+03
1	300.00	5.7946-03	1.0503-03	7.1168-04	9.0743-05	6.5803-06	1.1042+04
2	330.00	6.0412-03	1.1614-03	8.3773-04	1.1966-04	9.6770-06	1.2146+04

TABLE(III-2-9): The weights of fuel isotopes for 30%UO_2-70%THO2 fuel mixture and 12%U-235 enrichment level

a)

TIME STEP	TIME DAYS	K-EFFEKT.	CONVERS. RATIO	TH232 GRAMS	PA233 GRAMS	U233 GRAMS	BURNUP MWD/TO-U
0	.00	8.9006-01	0.0000	3.3774+00	0.0000	0.0000	0.0000
1	4.17	8.6540-01	1.4267-01	3.3770+00	3.8601-04	2.0662-05	1.5335+02
2	30.00	8.5113-01	3.7781-01	3.3744+00	2.0641-03	8.6670-04	1.1041+03
3	60.00	8.4248-01	4.8911-01	3.3713+00	3.0458-03	2.7626-03	2.2083+03
4	90.00	8.3568-01	5.4360-01	3.3681+00	3.5174-03	5.0725-03	3.3124+03
5	120.00	8.2969-01	5.7302-01	3.3650+00	3.7500-03	7.5112-03	4.4166+03
6	150.00	8.2409-01	5.9077-01	3.3618+00	3.8708-03	9.9480-03	5.5207+03
7	180.00	8.1869-01	6.0320-01	3.3586+00	3.9396-03	1.2324-02	6.6249+03
8	210.00	8.1342-01	6.1317-01	3.3554+00	3.9844-03	1.4616-02	7.7290+03
9	240.00	8.0823-01	6.2201-01	3.3521+00	4.0183-03	1.6814-02	8.8332+03
10	270.00	8.0312-01	6.3031-01	3.3489+00	4.0475-03	1.8915-02	9.9373+03
11	300.00	7.9808-01	6.3835-01	3.3456+00	4.0747-03	2.0923-02	1.1041+04
12	330.00	7.9311-01	6.4626-01	3.3423+00	4.1012-03	2.2839-02	1.2146+04

b)

TIME STEP	TIME DAYS	U234 GRAMS	U235 GRAMS	U236 GRAMS	U237 GRAMS	NP237 GRAMS	BURNUP MWD/TO-U
0	.00	0.0000	1.6854-01	0.0000	0.0000	0.0000	0.0000
1	4.17	5.8868-07	1.6760-01	1.9642-04	9.5152-08	1.3944-08	1.5335+02
2	30.00	2.7183-05	1.6187-01	1.3988-03	2.5841-06	3.1695-06	1.1041+03
3	60.00	9.5223-05	1.5537-01	2.7533-03	6.2870-06	1.6608-05	2.2083+03
4	90.00	1.9111-04	1.4967-01	4.0541-03	9.9692-06	4.0937-05	3.3124+03
5	120.00	3.0806-04	1.4299-01	5.2988-03	1.3550-05	7.5616-05	4.4166+03
6	150.00	4.4234-04	1.3713-01	6.4873-03	1.7019-05	1.2002-04	5.5207+03
7	180.00	5.9171-04	1.3149-01	7.6211-03	2.0376-05	1.7361-04	6.6249+03
8	210.00	7.5466-04	1.2605-01	8.7018-03	2.3621-05	2.3548-04	7.7290+03
9	240.00	9.3005-04	1.2081-01	9.7312-03	2.6760-05	3.0535-04	8.8332+03
10	270.00	1.1170-03	1.1577-01	1.0711-02	2.9793-05	3.8254-04	9.9373+03
11	300.00	1.3145-03	1.1091-01	1.1643-02	3.2724-05	4.6650-04	1.1041+04
12	330.00	1.5221-03	1.0624-01	1.2529-02	3.5554-05	5.5670-04	1.2146+04

c)

TIME STEP	TIME DAYS	POI51 GRAMS	POI52 GRAMS	POI53 GRAMS	U236 GRAMS	NP239 GRAMS	BURNUP MWD/TO-U
0	.00	0.0000	0.0000	0.0000	1.2360+00	0.0000	0.0000
1	4.17	0.0000	0.0000	0.0000	1.2359+00	7.6132-05	1.5335+02
2	30.00	0.0000	0.0000	0.0000	1.2349+00	1.0948-04	1.1041+03
3	60.00	0.0000	0.0000	0.0000	1.2338+00	1.1136-04	2.2083+03
4	90.00	0.0000	0.0000	0.0000	1.2327+00	1.1261-04	3.3124+03
5	120.00	0.0000	0.0000	0.0000	1.2315+00	1.1366-04	4.4166+03
6	150.00	0.0000	0.0000	0.0000	1.2304+00	1.1439-04	5.5207+03
7	180.00	0.0000	0.0000	0.0000	1.2292+00	1.1517-04	6.6249+03
8	210.00	0.0000	0.0000	0.0000	1.2281+00	1.1593-04	7.7290+03
9	240.00	0.0000	0.0000	0.0000	1.2269+00	1.1666-04	8.8332+03
10	270.00	0.0000	0.0000	0.0000	1.2257+00	1.1742-04	9.9373+03
11	300.00	0.0000	0.0000	0.0000	1.2245+00	1.1817-04	1.1041+04
12	330.00	0.0000	0.0000	0.0000	1.2233+00	1.1891-04	1.2146+04

d)

TIME STEP	TIME DAYS	PU239 GRAMS	PU240 GRAMS	PU241 GRAMS	PU242 GRAMS	AM243 GRAMS	BURNUP MWD/TO-U
0	.00	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000
1	4.17	5.6329-01	2.3347-07	0.0000	0.0000	0.0000	1.5335+02
2	30.00	6.1536-04	1.6737-05	1.3283-06	1.0935-08	0.0000	1.1041+03
3	60.00	1.6419-03	7.2837-05	1.0744-05	1.8339-07	0.0000	2.2083+03
4	90.00	2.3730-03	1.5132-04	3.3308-05	8.9155-07	1.3390-08	3.3124+03
5	120.00	3.0195-03	2.4557-04	7.1557-05	2.6298-06	5.9604-08	4.4166+03
6	150.00	3.5904-03	3.4927-04	1.2551-04	5.9196-06	1.7538-07	5.5207+03
7	180.00	4.0948-03	4.5815-04	1.9402-04	1.1251-05	4.1007-07	6.6249+03
8	210.00	4.5403-03	5.6917-04	2.7531-04	1.9056-05	8.2633-07	7.7290+03
9	240.00	4.9339-03	6.8018-04	3.6731-04	2.9691-05	1.4968-06	8.8332+03
10	270.00	5.2819-03	7.8971-04	4.6750-04	4.3437-05	2.5025-06	9.9373+03
11	300.00	5.5884-03	8.9671-04	5.7506-04	6.0499-05	3.9297-06	1.1041+04
12	330.00	5.8592-03	1.0006+03	6.8691-04	8.1016-05	5.8668-06	1.2146+04

SYSTEM-MATRIX A

Table(III-2-10):The weights of fuel isotopes for30%UO_2-70%THO_2 fuel mixture and 16%U-235 enrichment level

a)

ME EP	TIME DAYS	K-EFFEKT.	CONVERS. RATIO	TH232 GRAMS	PA233 GRAMS	U233 GRAMS	BURNUP MWD/TO-U
0	.00	9.7629-01	0.0000	3.3774+00	0.0000	0.0000	0.0000
1	4.17	9.5053-01	1.1548-01	3.3771+00	3.1897-04	1.7082-05	1.5336+02
2	30.00	9.3641-01	3.0711-01	3.3749+00	1.7077-03	7.1938-04	1.1042+03
3	60.00	9.2740-01	3.9880-01	3.3723+00	2.5199-03	2.3032-03	2.2083+03
4	90.00	9.2005-01	4.4382-01	3.3697+00	2.9115-03	4.2487-03	3.3125+03
5	120.00	9.1329-01	4.6793-01	3.3671+00	3.1068-03	6.3222-03	4.4167+03
6	150.00	9.0682-01	4.8251-01	3.3645+00	3.2107-03	8.4158-03	5.5208+03
7	180.00	9.0050-01	4.9276-01	3.3618+00	3.2722-03	1.0480-02	6.6250+03
8	210.00	8.9429-01	5.0106-01	3.3592+00	3.3143-03	1.2494-02	7.7292+03
9	240.00	8.8816-01	5.0849-01	3.3564+00	3.3475-03	1.4448-02	8.8333+03
10	270.00	8.8210-01	5.1557-01	3.3537+00	3.3768-03	1.6339-02	9.9375+03
11	300.00	8.7612-01	5.2250-01	3.3510+00	3.4046-03	1.8167-02	1.1042+04
12	330.00	8.7021-01	5.2939-01	3.3482+00	3.4318-03	1.9931-02	1.2146+04

b)

E P	TIME DAYS	U234 GRAMS	U235 GRAMS	U236 GRAMS	U237 GRAMS	NP237 GRAMS	BURNUP MWD/TO-U
	.00	0.0000	2.2471-01	0.0000	0.0000	0.0000	0.0000
	4.17	4.1736-07	2.2376-01	2.0770-04	8.7395-08	1.2962-08	1.5336+02
	30.00	1.9237-05	2.1791-01	1.4858-03	2.3797-06	2.9187-06	1.1042+03
	60.00	6.7240-05	2.1122-01	2.9387-03	5.8052-06	1.5345-05	2.2083+03
	90.00	1.5484-04	2.0467-01	4.3507-03	9.2462-06	3.7981-05	3.3125+03
	120.00	2.1743-04	1.9827-01	5.7195-03	1.2636-05	7.0494-05	4.4167+03
	150.00	3.1258-04	1.9202-01	7.0444-03	1.5967-05	1.1247-04	5.5208+03
	180.00	4.1893-04	1.8593-01	8.3258-03	1.9240-05	1.6349-04	6.6250+03
	210.00	5.3582-04	1.7998-01	9.5644-03	2.2452-05	2.2315-04	7.7292+03
	240.00	6.6202-04	1.7418-01	1.0761-02	2.5606-05	2.9104-04	8.8333+03
	270.00	7.9763-04	1.6853-01	1.1917-02	2.8701-05	3.6676-04	9.9375+03
	300.00	9.4201-04	1.6302-01	1.3032-02	3.1738-05	4.4993-04	1.1042+04
	330.00	1.0948-03	1.5764-01	1.4107-02	3.4716-05	5.4015-04	1.2146+04

c)

TIME STEP	TIME DAYS	PU239 GRAMS	PU240 GRAMS	PU241 GRAMS	PU242 GRAMS	AM243 GRAMS	BURNUP MWD/TO-U
0	.00	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000
1	4.17	4.6273-05	1.5354-07	0.0000	0.0000	0.0000	1.5336+02
2	30.00	6.7795-04	1.2087-05	7.2531-07	4.6148-09	0.0000	1.1042+03
3	60.00	1.3841-03	4.7795-05	5.9152-06	7.5843-08	0.0000	2.2083+03
4	90.00	2.0288-03	1.0129-04	1.8970-05	3.7540-07	4.7320-09	3.3125+03
5	120.00	2.6160-03	1.6761-04	4.1835-05	1.1323-06	2.1948-08	4.4167+03
6	150.00	3.1503-03	2.4291-04	7.5343-05	2.6092-06	6.5945-08	5.5208+03
7	180.00	3.6362-03	3.2427-04	1.1954-04	5.0777-06	1.5736-07	6.6250+03
8	210.00	4.0779-03	4.0951-04	1.7393-04	8.8015-06	3.2399-07	7.7292+03
9	240.00	4.4792-03	4.9694-04	2.3772-04	1.4027-05	5.9935-07	8.8333+03
10	270.00	4.8435-03	5.8531-04	3.0990-04	2.0975-05	1.0220-06	9.9375+03
11	300.00	5.1740-03	6.7365-04	3.8938-04	2.9839-05	1.6368-06	1.1042+04
12	330.00	5.4735-03	7.6121-04	4.7502-04	4.0783-05	2.4908-06	1.2146+04

d)

TIME STEP	TIME DAYS	POIS1 GRAMS	POIS2 GRAMS	POIS3 GRAMS	U238 GRAMS	NP239 GRAMS	BUNUP MWD/TO-U
0	.00	0.0000	0.0000	0.0000	1.1798+00	0.0000	0.0000
1	4.17	0.0000	0.0000	0.0000	1.1796+00	6.2444-05	1.5336+02
2	30.00	0.0000	0.0000	0.0000	1.1789+00	8.9764-05	1.1042+03
3	60.00	0.0000	0.0000	0.0000	1.1779+00	9.1159-05	2.2083+03
4	90.00	0.0000	0.0000	0.0000	1.1770+00	9.2120-05	3.3125+03
5	120.00	0.0000	0.0000	0.0000	1.1761+00	9.2914-05	4.4167+03
6	150.00	0.0000	0.0000	0.0000	1.1751+00	9.3648-05	5.5208+03
7	180.00	0.0000	0.0000	0.0000	1.1742+00	9.4359-05	6.6250+03
8	210.00	0.0000	0.0000	0.0000	1.1732+00	9.5062-05	7.7292+03
9	240.00	0.0000	0.0000	0.0000	1.1722+00	9.5764-05	8.8333+03
10	270.00	0.0000	0.0000	0.0000	1.1713+00	9.6468-05	9.9375+03
11	300.00	0.0000	0.0000	0.0000	1.1703+00	9.7176-05	1.1042+04
12	330.00	0.0000	0.0000	0.0000	1.1693+00	9.7886-05	1.2146+04

Table(III-2-11) The weights of fuel isotopes for 30%UO_2-70%THO_2 fuel mixture and 20%U-235 enrichment level

a)

TIME STEP	TIME DAYS	K-EFFECT.	CONVERS. RATIO	TH232 GRAMS	PA233 GRAMS	U233 GRAMS	BURNUP MWD/TO-U
0	.00	1.0375+00	0.0000	3.3774+00	0.0000	0.0000	0.0000
1	4.17	1.0113+00	9.7653-02	3.3771+00	2.7776-04	1.4878-05	1.5336+
2	30.00	9.9840-01	2.6194-01	3.3752+00	1.4878-03	6.2804-04	1.1042+
3	60.00	9.8945-01	3.4101-01	3.3730+00	2.1947-03	2.0160-03	2.2084+
4	90.00	9.8223-01	3.7971-01	3.3707+00	2.5354-03	3.7289-03	3.3126+
5	120.00	9.7553-01	4.0021-01	3.3685+00	2.7052-03	5.5645-03	4.4167+
6	150.00	9.6916-01	4.1239-01	3.3662+00	2.7956-03	7.4288-03	5.5209+
7	180.00	9.6270-01	4.2081-01	3.3638+00	2.8491-03	9.2789-03	6.6251+
8	210.00	9.5642-01	4.2700-01	3.3615+00	2.8858-03	1.1095-02	7.7293+
9	240.00	9.5021-01	4.3351-01	3.3592+00	2.9148-03	1.2670-02	8.8335+
10	270.00	9.4405-01	4.3921-01	3.3568+00	2.9404-03	1.4599-02	9.9377+
11	300.00	9.3794-01	4.4471-01	3.3544+00	2.9646-03	1.6281-02	1.1042+
12	330.00	9.3190-01	4.5052-01	3.3520+00	2.9884-03	1.7917-02	1.2146+

b)

TIME STEP	TIME DAYS	U234 GRAMS	U235 GRAMS	U236 GRAMS	U237 GRAMS	NP237 GRAMS	BURNUP MWD/TO-U
0	.00	0.0000	2.8057-01	0.0000	0.0000	0.0000	0.0000
1	4.17	3.2375-05	2.7990-01	2.1784-04	8.2447-08	0.0000	1.5336+
2	30.00	1.4886-04	2.7397-01	1.5623-03	2.2462-06	2.7420-06	1.1042+
3	60.00	5.1917-04	2.6714-01	3.0982-03	5.4851-06	1.4499-05	2.2084+
4	90.00	1.0358-03	2.6042-01	4.6008-03	8.7549-06	3.5983-05	3.3126+
5	120.00	1.6755-03	2.5383-01	6.0677-03	1.1994-05	6.6960-05	4.4167+
6	150.00	2.4082-03	2.4732-01	7.4981-03	1.5198-05	1.0712-04	5.5209+
7	180.00	3.2283-03	2.4095-01	8.8921-03	1.8366-05	1.5617-04	6.6251+
8	210.00	4.1298-03	2.3467-01	1.0250-02	2.1498-05	2.1379-04	7.7293+
9	240.00	5.1067-03	2.2851-01	1.1572-02	2.4594-05	2.7968-04	8.8335+
10	270.00	6.1619-03	2.2246-01	1.2859-02	2.7655-05	3.5354-04	9.9377+
11	300.00	7.2966-03	2.1654-01	1.4111-02	3.0679-05	4.3508-04	1.1042+
12	330.00	8.4804-03	2.1072-01	1.5329-02	3.3667-05	5.2399-04	1.2146+

c)

TIME STEP	TIME DAYS	POIS1 GRAMS	POIS2 GRAMS	POIS3 GRAMS	U238 GRAMS	NP239 GRAMS	BUNUP MWD/TO-U
0	.00	0.0000	0.0000	0.0000	1.1235+00	0.0000	0.0000
1	4.17	0.0000	0.0000	0.0000	1.1234+00	5.3264-05	1.5336+02
2	30.00	0.0000	0.0000	0.0000	1.1227+00	7.6494-05	1.1042+03
3	60.00	0.0000	0.0000	0.0000	1.1219+00	7.7563-05	2.2084+03
4	90.00	0.0000	0.0000	0.0000	1.1212+00	7.8325-05	3.3126+03
5	120.00	0.0000	0.0000	0.0000	1.1204+00	7.8956-05	4.4167+03
6	150.00	0.0000	0.0000	0.0000	1.1195+00	7.9548-05	5.5209+03
7	180.00	0.0000	0.0000	0.0000	1.1187+00	8.0127-05	6.6251+03
8	210.00	0.0000	0.0000	0.0000	1.1179+00	8.0704-05	7.7293+03
9	240.00	0.0000	0.0000	0.0000	1.1171+00	8.1283-05	8.8335+03
10	270.00	0.0000	0.0000	0.0000	1.1163+00	8.1865-05	9.9377+03
11	300.00	0.0000	0.0000	0.0000	1.1154+00	8.2451-05	1.1042+04
12	330.00	0.0000	0.0000	0.0000	1.1146+00	8.3041-05	1.2146+04

d)

TIME STEP	TIME DAYS	PU_{239} GRAMS	PU_{240} GRAMS	PU_{241} GRAMS	PU_{242} GRAMS	AM_{243} GRAMS	BURNUP MWD/TO-U
0	.00	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000
1	4.17	3.9508-05	1.1045-07	0.0000	0.0000	0.0000	1.5336+02
2	30.00	5.8287-04	8.4976-06	4.4558-07	0.0000	0.0000	1.1042+03
3	60.00	1.2013-03	3.3920-05	3.6759-06	3.4711-08	0.0000	2.2084+03
4	90.00	1.7748-03	7.2775-05	1.1962-05	1.8389-07	0.0000	3.3126+03
5	120.00	2.3079-03	1.2194-04	2.6803-05	5.6691-07	7.8530-09	4.4167+03
6	150.00	2.8022-03	1.7889-04	4.9069-05	1.3280-06	2.7788-08	5.5209+03
7	180.00	3.2601-03	2.4161-04	7.9138-05	2.6235-06	6.9771-08	6.6251+03
8	210.00	3.6842-03	3.0848-04	1.1703-04	4.6137-06	1.4800-07	7.7293+03
9	240.00	4.0766-03	3.7825-04	1.6249-04	7.4568-06	2.7888-07	8.8335+03
10	270.00	4.4395-03	4.4990-04	2.1510-04	1.1304-05	4.8326-07	9.9377+03
11	300.00	4.7749-03	5.2262-04	2.7431-04	1.6298-05	7.8420-07	1.1042+04
12	330.00	5.0845-03	5.9578-04	3.3950-04	2.2568-05	1.2087-06	1.2146+04

Table(III-2-12):The weights of fuel isotopes for 50%UO_2-50THO_2 fuel mixture and 10%U-235 enrichment level

a)

TIME STEP	TIME DAYS	K-EFFECT.	CO VERS. D T10	TH232 GRAMS	PA233 GRAMS	U233 GRAMS	BURNUP MWD/TON-U
0	.00	1.0100+00	0.0000	2.3276+06	0.0000	0.0000	0.0000
1	4.17	9.8352-01	1.0115-01	2.3274+06	2.1179-04	1.1363-05	1.5718+02
2	30.00	9.7050-01	3.2638-01	2.3260+06	1.1341-03	6.7203-04	1.1317+03
3	60.00	9.6199-01	3.8913-01	2.3243+06	1.6712-03	1.5309-03	2.2634+03
4	90.00	9.5412-01	4.2962-01	2.3225+06	1.9290-03	2.0247-03	3.3950+03
5	120.00	9.4660-01	4.3962-01	2.3208+06	2.0574-03	2.5051-03	4.5267+03
6	150.00	9.3931-01	4.5182-01	2.3191+06	2.1259-03	2.6010-03	5.6584+03
7	180.00	9.3206-01	4.6031-01	2.3173+06	2.1630-03	2.6004-03	6.7901+03
8	210.00	9.2483-01	4.6800-01	2.3155+06	2.1857-03	2.3290-03	7.9218+03
9	240.00	9.1781-01	4.7508-01	2.3137+06	2.2117-03	2.6408-03	9.0534+03
10	270.00	9.1097-01	4.8338-01	2.3119+06	2.2382-03	1.0014-02	1.0185+04
11	300.00	9.0405-01	4.9060-01	2.3101+06	2.2548-03	1.2147-02	1.1317+04
12	330.00	8.9734-01	4.9702-01	2.3083+06	2.2701-03	1.3341-02	1.2448+04

b)

TIME STEP	TIME DAYS	U234 GRAMS	U235 GRAMS	U236 GRAMS	U237 GRAMS	NP237 GRAMS	BURNUP MWD/TON-U
0	.00	0.0000	2.3[illegible]-01	0.0000	0.0000	0.0000	0.0000
1	4.17	2.6422[illegible]	2.2[illegible]-01	2.0[illegible]-[illegible]	[illegible].2117-[illegible]	1.0[illegible]-[illegible]	1.5718+[illegible]
2	30.00	1.2187-05	2.3696-01	1.4752-07	2.2369-06	2.7439-06	1.1317+03
3	60.00	[illegible].0598-05	2.3[illegible]-01	2.[illegible]156-[illegible]	[illegible].4404-06	1.[illegible]-05	2.2634+03
4	90.00	[illegible].5430-05	2.1367-01	4.3157-07	[illegible].6705-06	3.5609-05	3.3950+03
5	120.00	1.3785-04	2.0755-01	5.6765-07	1.1975-05	6.[illegible]-05	4.5267+03
6	150.00	1.9837-04	2.0138-01	6.9020-07	1.5032-05	1.0509-04	5.6584+03
7	180.00	2.6613-04	1.9535-01	8.[illegible]-[illegible]	1.8151-05	1.5413-04	6.7901+03
8	210.00	3.4078-04	1.8965-01	9.5004-07	2.1231-05	2.1069-04	7.9218+03
9	240.00	[illegible].2183-04	1.8370-01	1.0705-06	2.4232-05	2.7526-04	9.0534+03
10	270.00	[illegible].0397-04	1.7859-01	1.1965-06	2.7232-05	3.4789-04	1.0185+04
11	300.00	[illegible].9201-04	1.7259-01	1.[illegible]-[illegible]	3.0230-05	[illegible].2764-04	1.1317+04
12	330.00	[illegible].0071-04	1.6702-01	1.[illegible]-[illegible]	3.3146-05	5.1361-04	1.2448+04

c)

IME TEP	TIME DAYS	POIS1 GRAMS	P IS2 GR MS	POIS3 GRAMS	U238 GRAMS	NP239 GRAMS	BURNUP MWD/TO-U
0	.00	0.0000	0. 000	0.0000	2.1043+00	0.0000	0.0000
1	4.17	0.0000	0. 000	0.0000	2.1041+00	1.0676-04	1.5718+02
2	30.00	0.0000	0. 000	0.0000	2.1027+00	1.5347-04	1.1317+03
3	60.00	0.0000	0.0000	0.0000	2.1012+00	1.5563-04	2.2634+03
4	90.00	0.0000	0.0000	0.0000	2.0996+00	1.5723-04	3.3950+03
5	120.00	0.0000	0.0000	0.0000	2.0980+00	1.5867-04	4.5267+03
6	150.00	0.0000	0.0000	0.0000	2.0964+00	1.6008-04	5.6584+03
7	180.00	0.0000	0.0000	0.0000	2.0947+00	1.6147-04	6.7901+03
8	210.00	0.0000	0.0000	0.0000	2.0931+00	1.6287-04	7.9218+03
9	240.00	0.0000	0.0000	0.0000	2.0914+00	1.6427-04	9.0534+03
10	270.00	0.0000	0.0000	0.0000	2.0898+00	1.6567-04	1.0185+04
11	300.00	0.0000	0.0000	0.0000	2.0881+00	1.6708-04	1.1317+04
12	330.00	0.0000	0.0000	0.0000	2.0864+00	1.6849-04	1.2448+04

d)

ME TEP	TIME DAYS	PU239 GRAMS	U240 RAMS	PU241 GRAMS	PU242 GRAMS	AM243 GRAMS	BURNUP MWD/TO-U
0	.00	0.0000	0. 000	0.0000	0.0000	0.0000	0.0000
1	4.17	7.9128-05	2. 033-07	0.0000	0.0000	0.0000	1.5718+02
2	30.00	1.1616-03	1. 743-05	1.1265-06	5.4173-09	0.0000	1.1317+03
3	60.00	2.3765-03	7.3033-05	9.1991-06	1.0975-07	0.0000	2.2634+03
4	90.00	3.4929-03	1.6554-04	2.9451-05	5.4795-07	7.0476-09	3.3950+03
5	120.00	4.5179-03	2.7457-04	6.4901-05	1.6407-06	3.0607-08	4.5267+03
6	150.00	5.4591-03	3.9932-04	1.1681-04	3.7941-06	9.2021-08	5.6584+03
7	180.00	6.3239-03	5.3541-04	1.8526-04	7.3708-06	2.1968-07	6.7901+03
8	210.00	7.1189-03	6.7960-04	2.6953-04	1.2757-05	4.5266-07	7.9218+03
9	240.00	7.8483-03	8.2950-04	3.6994-04	2.0306-05	8.3625-07	9.0534+03
0	270.00	8.5190-03	9.8293-04	4.8055-04	3.0335-05	1.4264-06	1.0185+04
1	300.00	9.1350-03	1.1397-03	6.0429-04	4.3195-05	2.2958-06	1.1317+04
2	330.00	9.7006-03	1.2956-03	7.3805-04	5.8918-05	3.4821-06	1.2448+04

Table(III-2-13):The weights of fuel isotopes for 50%UO_2-50%THO_2 fuel mixture and 12%U-235 enrichment level

a)

TIME STEP	TIME DAYS	K-EFFECT.	CONVERS. RATIO	Th_{232} GRAMS	Pa_{233} GRAMS	U_{233} GRAMS	BURNUP MWD/TO-U
0	.00	1.0529+00	0.0000	2.3276+00	0.0000	0.0000	0.0000
1	4.17	1.0316+00	1.4308-01	2.3274+00	1.8955-04	1.0153-05	1.5721+02
2	30.00	1.0119+00	2.8915-01	2.3261+00	1.0154-03	4.2869-04	1.1319+03
3	60.00	1.0101+00	3.4487-01	2.3246+00	1.4965-03	1.3756-03	2.2638+03
4	90.00	1.0026+00	3.7294-01	2.3231+00	1.7276-03	2.5444-03	3.3958+03
5	120.00	9.9524-01	3.8865-01	2.3215+00	1.8426-03	3.7966-03	4.5277+03
6	150.00	9.8801-01	3.9895-01	2.3200+00	1.9041-03	5.0690-03	5.6596+03
7	180.00	9.8065-01	4.0677-01	2.3184+00	1.9408-03	6.3324-03	6.7915+03
8	210.00	9.7327-01	4.1305-01	2.3168+00	1.9664-03	7.5738-03	7.9234+03
9	240.00	9.6676-01	4.1989-01	2.3152+00	1.9868-03	8.7876-03	9.0554+03
10	270.00	9.5983-01	4.2607-01	2.3136+00	2.0050-03	9.9714-03	1.0187+04
11	300.00	9.5301-01	4.3220-01	2.3120+00	2.0224-03	1.1125-02	1.1319+04
12	330.00	9.4628-01	4.3831-01	2.3103+00	2.0394-03	1.2247-02	1.2451+04

b)

TIME STEP	TIME DAYS	U234 GRAMS	U235 GRAMS	U236 GRAMS	U237 GRAMS	NP237 GRAMS	BURNUP MWD/TO-U
0	.00	0.0000	2.8047-01	0.0000	0.0000	0.0000	0.0000
1	4.17	2.1587-07	2.7951-01	2.1459-04	7.8648-08	0.0000	1.5721+02
2	30.00	9.9303-06	2.7364-01	1.5389-03	2.1432-06	2.6175-06	1.1319+03
3	60.00	3.4646-05	2.6691-01	3.0491-03	5.2275-06	1.3829-05	2.2638+03
4	90.00	6.9430-05	2.6029-01	4.5250-03	8.3418-06	3.4309-05	3.3958+03
5	120.00	1.1196-04	2.5379-01	5.9655-03	1.1434-05	6.3657-05	4.5277+03
6	150.00	1.6107-04	2.4740-01	7.3705-03	1.4503-05	1.0221-04	5.6596+03
7	180.00	2.1614-04	2.4114-01	8.7403-03	1.7547-05	1.4910-04	6.7915+03
8	210.00	2.7679-04	2.3498-01	1.0075-02	2.0568-05	2.0427-04	7.9234+03
9	240.00	3.4277-04	2.2894-01	1.1376-02	2.3565-05	2.6746-04	9.0554+03
10	270.00	4.1388-04	2.2300-01	1.2643-02	2.6536-05	3.3842-04	1.0187+04
11	300.00	4.8995-04	2.1718-01	1.3876-02	2.9481-05	4.1686-04	1.1319+04
12	330.00	5.7063-04	2.1145-01	1.5076-02	3.2399-05	5.0245-04	1.2451+04

c)

TIME STEP	TIME DAYS	POIS1 GRAMS	POIS2 GRAMS	POIS3 GRAMS	U238 GRAMS	NP239 GRAMS	BUNUP MWD/TO-U
0	.00	0.0000	0.0000	0.0000	2.0565+00	0.0000	0.0000
1	4.17	0.0000	0.0000	0.0000	2.0563+00	9.5612-05	1.5721+02
2	30.00	0.0000	0.0000	0.0000	2.0551+00	1.3734-04	1.1319+03
3	60.00	0.0000	0.0000	0.0000	2.0537+00	1.3914-04	2.2638+03
4	90.00	0.0000	0.0000	0.0000	2.0523+00	1.4051-04	3.3958+03
5	120.00	0.0000	0.0000	0.0000	2.0509+00	1.4172-04	4.5277+03
6	150.00	0.0000	0.0000	0.0000	2.0494+00	1.4290-04	5.6596+03
7	180.00	0.0000	0.0000	0.0000	2.0480+00	1.4408-04	6.7915+03
8	210.00	0.0000	0.0000	0.0000	2.0465+00	1.4526-04	7.9234+03
9	240.00	0.0000	0.0000	0.0000	2.0450+00	1.4646-04	9.0554+03
10	270.00	0.0000	0.0000	0.0000	2.0435+00	1.4765-04	1.0187+04
11	300.00	0.0000	0.0000	0.0000	2.0420+00	1.4886-04	1.1319+04
12	330.00	0.0000	0.0000	0.0000	2.0405+00	1.5007-04	1.2451+04

d)

TIME STEP	TIME DAYS	PU239 GRAMS	PU240 GRAMS	PU241 GRAMS	PU242 GRAMS	AM243 GRAMS	BURNUP MWD/TO-U
0	.00	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000
1	4.17	7.0922-05	1.9537-07	0.0000	0.0000	0.0000	1.5721+02
2	30.00	1.0468-03	1.5103-05	7.7727-07	0.0000	0.0000	1.1319+03
3	60.00	2.1564-03	6.0167-05	6.3776-06	5.9591-08	0.0000	2.2638+03
4	90.00	3.1909-03	1.2891-04	2.0686-05	3.1300-07	0.0000	3.3958+03
5	120.00	4.1541-03	2.1591-04	4.6195-05	9.6128-07	1.2686-08	4.5277+03
6	150.00	5.0509-03	3.1690-04	8.4276-05	2.2419-06	4.5910-08	5.6596+03
7	180.00	5.8859-03	4.2855-04	1.3546-04	4.4093-06	1.1558-07	6.7915+03
8	210.00	6.6634-03	5.4828-04	1.9965-04	7.7221-06	2.4425-07	7.9234+03
9	240.00	7.3873-03	6.7406-04	2.7638-04	1.2432-05	4.5930-07	9.0554+03
10	270.00	8.0611-03	8.0430-04	3.6486-04	1.8778-05	7.9376-07	1.0187+04
11	300.00	8.6882-03	9.3774-04	4.6416-04	2.6980-05	1.2867-06	1.1319+04
12	330.00	9.2714-03	1.0733-03	5.7325-04	3.7238-05	1.9780-06	1.2451+04

Table(III-2-14):The weights of fuel isotopes for 50%UO_2-50%THO_2 fuel mixture and 16%U-235 enrichment level

a)

ME EP	TIME DAYS	K-EFFEKT.	CONVERS. RATIO	TH232 GRAMS	PA233 GRAMS	U233 GRAMS	BURNUP MWD/TO-U
0	.00	1.1260+00	0.0000	2.3276+00	0.0000	0.0000	0.0000
1	4.17	1.1015+00	1.1774-01	2.3275+00	1.6086-04	8.6189-06	1.5723+02
2	30.00	1.0898+00	2.3858-01	2.3264+00	8.6167-04	3.6457-04	1.1320+03
3	60.00	1.0811+00	2.8489-01	2.3251+00	1.2696-03	1.[illegible]51-03	2.2640+03
4	90.00	1.0740+00	3.0802-01	2.3238+00	1.4655-03	2.1755-03	3.3961+03
5	120.00	1.0675+00	3.2065-01	2.3224+00	1.5626-03	3.2555-03	4.5281+03
6	150.00	1.0607+00	3.2855-01	2.3211+00	1.6140-03	4.3592-03	5.6601+03
7	180.00	1.0541+00	3.3436-01	2.3198+00	1.6442-03	5.4617-03	6.7921+03
8	210.00	1.0476+00	3.3927-01	2.3184+00	1.6648-03	6.5517-03	7.9242+03
9	240.00	1.0411+00	3.4382-01	2.3171+00	1.6810-03	7.6240-03	9.0562+03
0	270.00	1.0346+00	3.4825-01	2.3157+00	1.6952-03	8.6765-03	1.0188+04
1	300.00	1.0282+00	3.5266-01	2.3143+00	1.7086-03	9.7084-03	1.1320+04
2	330.00	1.0219+00	3.5708-01	2.3130+00	1.7217-03	1.0719-02	1.2452+04

b)

ME EP	TIME DAYS	U234 GRAMS	U235 GRAMS	U236 GRAMS	U237 GRAMS	NP237 GRAMS	BURNUP MWD/TO-U
0	.00	0.0000	3.7388-01	0.0000	0.0000	0.0000	0.0000
1	4.17	1.5859-07	3.7291-01	2.2959-04	7.3731-08	0.0000	1.5723+02
2	30.00	7.2531-06	3.6691-01	1.6503-03	2.0071-06	2.4580-06	1.1320+03
3	60.00	2.5231-05	3.6000-01	3.2789-03	4.8993-06	1.2982-05	2.2640+03
4	90.00	5.0469-05	3.5317-01	4.8811-03	7.8338-06	3.2268-05	3.3961+03
5	120.00	8.1275-05	3.4642-01	6.4554-03	1.0759-05	6.0187-05	4.5281+03
6	150.00	1.1681-04	3.3975-01	8.0014-03	1.3673-05	9.6559-05	5.6601+03
7	180.00	1.5666-04	3.3316-01	9.5191-03	1.6577-05	1.4120-04	6.7921+03
8	210.00	2.0056-04	3.2664-01	1.1009-02	1.9470-05	1.9392-04	7.9242+03
9	240.00	2.4837-04	3.2021-01	1.2470-02	2.2354-05	2.5455-04	9.0562+03
0	270.00	2.9997-04	3.1386-01	1.3904-02	2.5227-05	3.2291-04	1.0188+04
1	300.00	3.5528-04	3.0758-01	1.5310-02	2.8089-05	3.9880-04	1.1320+04
2	330.00	4.1420-04	3.0138-01	1.6689-02	3.0940-05	4.8205-04	1.2452+04

c)

IME TEP	TIME DAYS	POIS1 GRAMS	POIS2 GRAMS	POIS3 GRAMS	U238 GRAMS	NP239 GRAMS	BUNUP MWD/TO-U
0	.00	0.0000	0.0000	0.0000	1.9627+00	0.0000	0.0000
1	4.17	0.0000	0.0000	0.0000	1.9625+00	7.9983-05	1.5723+02
2	30.00	0.0000	0.0000	0.0000	1.9615+00	1.1464-04	1.1320+03
3	60.00	0.0000	0.0000	0.0000	1.9603+00	1.1595-04	2.2640+03
4	90.00	0.0000	0.0000	0.0000	1.9591+00	1.1699-04	3.3961+03
5	120.00	0.0000	0.0000	0.0000	1.9579+00	1.1788-04	4.5281+03
6	150.00	0.0000	0.0000	0.0000	1.9567+00	1.1874-04	5.6601+03
7	180.00	0.0000	0.0000	0.0000	1.9555+00	1.1959-04	6.7921+03
8	210.00	0.0000	0.0000	0.0000	1.9543+00	1.2045-04	7.9242+03
9	240.00	0.0000	0.0000	0.0000	1.9530+00	1.2132-04	9.0562+03
10	270.00	0.0000	0.0000	0.0000	1.9518+00	1.2219-04	1.0188+04
11	300.00	0.0000	0.0000	0.0000	1.9505+00	1.2307-04	1.1320+04
12	330.00	0.0000	0.0000	0.0000	1.9492+00	1.2396-04	1.2452+04

d)

ME EP	TIME DAYS	PU239 GRAMS	PU240 GRAMS	PU241 GRAMS	PU242 GRAMS	AM243 GRAMS	BURNUP MWD/TO-U
	.00	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000
	4.17	5.9387-05	1.3144-07	0.0000	0.0000	0.0000	1.5723+02
	30.00	8.8200-04	9.7463-06	4.2360-07	0.0000	0.0000	1.1320+03
	60.00	1.8328-03	3.9167-05	3.4819-06	2.4478-08	0.0000	2.2640+03
	90.00	2.7359-03	8.5002-05	1.1458-05	1.2781-07	0.0000	3.3961+03
	120.00	3.5926-03	1.4427-04	2.6023-05	3.9466-07	0.0000	4.5281+03
	150.00	4.4047-03	2.1448-04	4.8322-05	9.3339-07	1.2283-08	5.6601+03
	180.00	5.1745-03	2.9358-04	7.9065-05	1.8632-06	3.7654-08	6.7921+03
	210.00	5.9041-03	3.7988-04	1.1862-04	3.3120-06	8.5981-08	7.9242+03
	240.00	6.5953-03	4.7201-04	1.6706-04	5.4123-06	1.6733-07	9.0562+03
	270.00	7.2502-03	5.6882-04	2.2428-04	8.2940-06	2.9540-07	1.0188+04
	300.00	7.8704-03	6.6940-04	2.9002-04	1.2087-05	4.8628-07	1.1320+04
	330.00	8.4576-03	7.7295-04	3.6388-04	1.6916-05	7.5913-07	1.2452+04

SYSTEM-MATRIX A

Table(III-2-15):The weights of fuel isotopes for 50%UO_2-50%THO_2 fuel mixture and 20%U-235 enrichment level

a)

TIME STEP	TIME DAYS	K-EFFEKT.	CONVERS. RATIO	TH232 GRAMS	PA233 GRAMS	U233 GRAMS	BURNUP MWD/TO-U
0	.00	1.1733+00	0.0000	2.3976+00	0.0000	0.0000	0.0000
1	4.17	1.1512+00	1.0239-01	2.3975+00	1.4277-04	-.6509-06	1.5722+02
2	30.00	1.1406+00	2.0457-01	2.3965+00	-.6428-04	3.0383-04	1.1320+03
3	60.00	1.1322+00	2.4403-01	2.3954+00	1.1256-03	1.0436-03	2.2640+03
4	90.00	1.1254+00	2.6480-01	2.3942+00	1.2000-03	1.0385-03	3.3960+03
5	120.00	1.1192+00	2.7547-01	2.3930+00	1.3846-03	2.0058-03	4.5280+03
6	150.00	1.1132+00	2.8102-01	2.3919+00	1.4204-03	3.0978-03	5.6600+03
7	180.00	1.1073+00	2.8648-01	2.3907+00	1.4553-03	4.0923-03	6.7920+03
8	210.00	1.1013+00	2.9022-01	2.3895+00	1.4725-03	5.0792-03	7.9240+03
9	240.00	1.0954+00	2.9363-01	2.3883+00	1.4858-03	6.0536-03	9.0560+03
10	270.00	1.0895+00	2.9695-01	2.3871+00	1.4972-03	7.0137-03	1.0188+04
11	300.00	1.0837+00	3.0020-01	2.3859+00	1.5079-03	7.9585-03	1.1320+04
12	330.00	1.0779+00	3.0350-01	2.3846+00	1.5183-03	8.6876-03	1.2452+04

b)

TIME STEP	TIME DAYS	U238 GRAMS	U239 GRAMS	U236 GRAMS	U237 GRAMS	NP237 GRAMS	BURNUP MWD/TO-U
0	.00	0.0000	4.6733-01	0.0000	0.0000	0.0000	0.0000
1	4.17	1.2484-07	4.6634-01	2.4051-08	7.0020-08	0.0000	1.5722+02
2	30.00	5.7003-06	4.6624-01	1.7440-03	1.9002-06	0.3378-06	1.1320+03
3	60.00	1.9784-05	4.6519-01	3.4723-03	0.6428-06	1.0314-05	2.2640+03
4	90.00	3.9513-05	4.6401-01	5.1781-03	7.4310-06	3.0627-05	3.3960+03
5	120.00	6.3560-05	4.6298-01	6.8607-03	1.0217-05	5.7223-05	4.5280+03
6	150.00	9.1275-05	4.6201-01	8.5196-03	1.2998-05	8.1960-05	5.6600+03
7	180.00	1.2253-04	4.6164-01	1.0155-02	1.5773-05	1.3867-04	6.7920+03
8	210.00	1.5654-04	4.1894-01	1.1766-02	1.8544-05	1.8522-04	7.9240+03
9	240.00	1.9380-04	4.1815-01	1.3353-02	2.1311-05	2.4347-04	9.0560+03
10	270.00	2.3004-04	4.6552-01	1.4916-02	2.4074-05	3.0929-04	1.0188+04
11	300.00	2.7719-04	4.6305-01	1.6456-02	2.6832-05	3.8256-04	1.1320+04
12	330.00	3.2320-04	4.6204-01	1.7972-02	2.9587-05	4.6309-04	1.2452+04

c)

TIME STEP	TIME DAYS	PU239 GRAMS	U240 GRAMS	PU241 GRAMS	PU242 GRAMS	AM243 GRAMS	BURNUP MWD/TO-U
0	.00	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000
1	4.17	5.1229-05	9.5424-08	0.0000	0.0000	0.0000	1.5722+02
2	30.00	7.6318-04	8.7327-06	2.5056-07	0.0000	0.0000	1.1320+03
3	60.00	1.5042-03	2.7403-05	2.0056-06	0.0239-09	0.0000	2.2640+03
4	90.00	2.3029-03	5.0451-05	6.9574-06	5.8196-08	0.0000	3.3960+03
5	120.00	3.1595-03	1.0265-04	1.5075-05	1.8559-07	0.0000	4.5280+03
6	150.00	3.8946-03	1.5305-04	3.0000-05	4.4603-07	0.0000	5.6600+03
7	180.00	4.5094-03	2.1251-04	4.9657-05	9.0169-07	0.6553-09	6.7920+03
8	210.00	5.2749-03	2.7720-04	7.5370-05	1.6102-06	2.9801-08	7.9240+03
9	240.00	5.9222-03	3.4705-04	1.0739-04	2.6740-06	6.4838-08	9.0560+03
10	270.00	6.5424-03	4.2107-04	1.4581-04	4.1364-06	1.2082-07	1.0188+04
11	300.00	7.1364-03	4.9913-04	1.9066-04	6.0853-06	2.0619-07	1.1320+04
12	330.00	7.7053-03	5.8007-04	2.4184-04	8.5065-06	3.2942-07	1.2452+04

1 SYSTEM-MATRIX A

d)

TIME STEP	TIME DAYS	PU251 GRAMS	U252 GRAMS	PU253 GRAMS	U239 GRAMS	NP239 GRAMS	BURNUP MWD/TO-U
0	.00	0.0000	0.0000	0.0000	1.8604+00	0.0000	0.0000
1	4.17	0.0000	0.0000	0.0000	1.8602+00	6.8952-05	1.5722+02
2	30.00	0.0000	0.0000	0.0000	1.8603+00	0.8609-05	1.1320+03
3	60.00	0.0000	0.0000	0.0000	1.8603+00	0.8592-05	2.2640+03
4	90.00	0.0000	0.0000	0.0000	1.8663+00	1.0042-04	3.3960+03
5	120.00	0.0000	0.0000	0.0000	1.8652+00	1.0111-04	4.5280+03
6	150.00	0.0000	0.0000	0.0000	1.8642+00	1.0177-04	5.6600+03
7	180.00	0.0000	0.0000	0.0000	1.8631+00	1.0241-04	6.7920+03
8	210.00	0.0000	0.0000	0.0000	1.8621+00	1.0305-04	7.9240+03
9	240.00	0.0000	0.0000	0.0000	1.8610+00	1.0370-04	9.0560+03
10	270.00	0.0000	0.0000	0.0000	1.8599+00	1.0436-04	1.0188+04
11	300.00	0.0000	0.0000	0.0000	1.8588+00	1.0503-04	1.1320+04
12	330.00	0.0000	0.0000	0.0000	1.8578+00	1.0569-04	1.2452+04

Table(III-2-16): The weights of fuel isotopes for 70%UO_2-30%THO_2 fuel mixture and 10%U-235 enrichment level

a)

TIME STEP	TIME DAYS	K-EFFECT.	CONVERS. RATIO	TH232 GRAMS	PA233 GRAMS	U233 GRAMS	BURNUP MWD/TO-
0	.00	1.1085+00	0.0000	1.4004+00	0.0000	0.0000	0.0000
1	4.17	1.0[illegible]+00	1.7169-01	1.4002+00	1.0335-04	5.5368-06	1.5677
2	30.00	1.0[illegible]+00	2.9193-01	1.3995+00	5.5371-04	2.3407-04	1.1287
3	60.00	1.0622+00	3.2381-01	1.3987+00	8.1540-04	7.5231-04	2.2575
4	90.00	1.0516+00	3.4071-01	1.3979+00	9.4082-04	1.3934-03	3.3862
5	120.00	1.0472+00	3.5101-01	1.3970+00	1.0030-03	2.0825-03	4.5149
6	150.00	1.0391+00	3.5848-01	1.3962+00	1.0361-03	2.7852-03	5.6436
7	180.00	1.0323+00	3.6477-01	1.3953+00	1.0559-03	3.4858-03	6.7724
8	210.00	1.0259+00	3.7058-01	1.3945+00	1.0695-03	4.1769-03	7.9011
9	240.00	1.0196+00	3.7622-01	1.3936+00	1.0805-03	4.8556-03	9.0298
10	270.00	1.0139+00	3.8182-01	1.3927+00	1.0902-03	5.5204-03	1.0159
11	300.00	1.0084+00	3.8741-01	1.3918+00	1.0995-03	6.1709-03	1.1287
12	330.00	9.9420-01	3.9301-01	1.3909+00	1.1085-03	6.8070-03	1.2416

b)

TIME STEP	TIME DAYS	U234 GRAMS	U235 GRAMS	U236 GRAMS	U237 GRAMS	NP237 GRAMS	BURNUP MWD/TO-[illegible]
0	.00	0.0000	3.2774-01	0.0000	0.0000	0.0000	0.0000
1	4.17	1.0650-07	3.2679-01	2.1852-04	7.2840-08	0.0000	1.5677+
2	30.00	4.8907-06	3.2091-01	1.5764-03	1.9844-06	2.4274-06	1.1287+
3	60.00	1.7000-05	3.1417-01	3.1263-03	4.8389-06	1.2818-05	2.2575+
4	90.00	3.4109-05	3.0753-01	4.6457-03	7.7293-06	3.1832-05	3.3862+
5	120.00	5.5091-05	3.0099-01	6.1539-03	1.0610-05	5.9330-05	4.5149+
6	150.00	7.9221-05	2.9455-01	7.5912-03	1.3481-05	9.5119-05	5.6436+
7	180.00	1.0642-04	2.8821-01	9.0178-03	1.6344-05	1.3901-04	6.7724+
8	210.00	1.3436-04	2.8196-01	1.0414-02	1.9199-05	1.9083-04	7.9011+
9	240.00	1.6291-04	2.7581-01	1.1781-02	2.2044-05	2.5037-04	9.0298+
10	270.00	2.0924-04	2.6974-01	1.3118-02	2.4879-05	3.1745-04	1.0159+
11	300.00	2.4207-04	2.6377-01	1.4426-02	2.7703-05	3.9189-04	1.1287+
12	330.00	2.8216-04	2.5789-01	1.5704-02	3.0514-05	4.7348-04	1.2416+

c)

TIME STEP	TIME DAYS	PU239 GRAMS	PU240 GRAMS	PU241 GRAMS	PU242 GRAMS	AM243 GRAMS	BURNUP MWD/TO-U
0	.00	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000
1	4.17	9.31[illegible]-05	2.2487-07	0.0000	0.0000	0.0000	1.5677+02
2	30.00	1.38[illegible]-03	1.7076-05	7.8166-07	0.0000	0.0000	1.1287+03
3	60.00	2.96[illegible]-03	6.8264-05	6.4563-06	5.0161-08	0.0000	2.2575+03
4	90.00	4.25[illegible]-03	1.4712-04	2.1077-05	2.6545-07	0.0000	3.3862+03
5	120.00	5.67[illegible]-03	2.4809-04	4.7395-05	8.1902-07	0.0000	4.5149+03
6	150.00	6.31[illegible]-03	3.6672-04	8.7094-05	1.9171-06	2.5774-08	5.6436+03
7	180.00	7.98[illegible]-03	4.9953-04	1.4100-04	3.7861-06	8.1752-08	6.7724+03
8	210.00	9.09[illegible]-03	6.4377-04	2.0931-04	6.6601-06	1.8324-07	7.9011+03
9	240.00	1.01[illegible]-02	7.9727-04	2.9178-04	1.0772-05	3.5198-07	9.0298+03
10	270.00	1.11[illegible]-02	9.5831-04	3.8786-04	1.6345-05	6.1938-07	1.0159+04
11	300.00	1.20[illegible]-02	1.1255-03	4.9681-04	2.3593-05	1.0128-06	1.1287+04
12	330.00	1.29[illegible]-02	1.2976-03	6.1773-04	3.2715-05	1.5714-06	1.2416+04

1 SYSTEM=MATRIX A

d)

TIME STEP	TIME DAYS	POIS1 GRAMS	POIS2 GRAMS	POIS3 GRAMS	U238 GRAMS	NP239 GRAMS	BUNUP MWD/TO-U
0	.00	0.0000	0.0000	0.0000	2.9497+00	0.0000	0.0000
1	4.17	0.0000	0.0000	0.0000	2.9495+00	1.2555-04	1.5677+02
2	30.00	0.0000	0.0000	0.0000	2.9479+00	1.8017-04	1.1287+03
3	60.00	0.0000	0.0000	0.0000	2.9460+00	1.8228-04	2.2575+03
4	90.00	0.0000	0.0000	0.0000	2.9442+00	1.8398-04	3.3862+03
5	120.00	0.0000	0.0000	0.0000	2.9423+00	1.8554-04	4.5149+03
6	150.00	0.0000	0.0000	0.0000	2.9404+00	1.8708-04	5.6436+03
7	180.00	0.0000	0.0000	0.0000	2.9385+00	1.8864-04	6.7724+03
8	210.00	0.0000	0.0000	0.0000	2.9365+00	1.9021-04	7.9011+03
9	240.00	0.0000	0.0000	0.0000	2.9346+00	1.9180-04	9.0298+03
10	270.00	0.0000	0.0000	0.0000	2.9326+00	1.9340-04	1.0159+04
11	300.00	0.0000	0.0000	0.0000	2.9306+00	1.9501-04	1.1287+04
12	330.00	0.0000	0.0000	0.0000	2.9286+00	1.9662-04	1.2416+04

Table(III-2-17):The weights of fuel isotopes for 70%UO_2 _30%THO_2 fuel mixture and 12%U-235 enrichment level

a)

TIME STEP	TIME DAYS	K-EFFEKT.	CO VERS. RATIO	TH232 GRAMS	PA233 GRAMS	U233 GRAMS	BURNUP MWD/TO-U
0	.00	1.1481+00	0.0000	1.4004+00	0.0000	0.0000	0.0000
1	4.17	1.1238+00	1.5297-01	1.4002+00	0.3507-05	5.0103-06	1.5677+02
2	30.00	1.1127+00	2.6006-01	1.3996+00	5.0006-04	0.1200-04	1.1288+03
3	60.00	1.1042+00	2.8841-01	1.3989+00	7.3756-04	6.8240-04	2.2575+03
4	90.00	1.0970+00	3.0329-01	1.3981+00	9.5103-04	1.2660-03	3.3863+03
5	120.00	1.0900+00	3.1212-01	1.3973+00	0.0721-04	1.8953-03	4.5150+03
6	150.00	1.0831+00	3.1832-01	1.3966+00	0.3600-04	2.5391-03	5.6438+03
7	180.00	1.0761+00	3.2342-01	1.3958+00	0.5838-04	3.1830-03	6.7725+03
8	210.00	1.0692+00	3.2809-01	1.3950+00	0.6630-04	3.8205-03	7.9013+03
9	240.00	1.0623+00	3.3262-01	1.3942+00	0.7569-04	4.4486-03	9.0300+03
10	270.00	1.0555+00	3.3712-01	1.3934+00	0.8307-04	5.0661-03	1.0159+04
11	300.00	1.0488+00	3.4163-01	1.3926+00	0.9179-04	5.6723-03	1.1288+04
12	330.00	1.0421+00	3.4618-01	1.3918+00	0.9844-04	6.2673-03	1.2416+04

b)

TIME STEP	TIME DAYS	U234 GRAMS	U235 GRAMS	U236 GRAMS	U237 GRAMS	NP237 GRAMS	BURNUP MWD/TO-U
0	.00	0.0000	3.1328-01	0.0000	0.0000	0.0000	0.0000
1	4.17	0.8123-09	3.0231-01	0.2053-04	7.0134-08	0.0000	1.5677+02
2	30.00	0.0300-06	3.0635-01	1.6503-03	1.9005-06	0.3299-06	1.1288+03
3	60.00	1.4017-05	3.0949-01	3.2785-03	0.6559-06	1.2336-05	2.2575+03
4	90.00	2.8040-05	3.0971-01	4.8008-03	7.4458-06	2.0679-05	3.3863+03
5	120.00	4.5169-05	3.0601-01	6.4564-03	1.0232-05	5.7260-05	4.5150+03
6	150.00	6.4947-05	3.0539-01	8.0852-03	1.3013-05	0.1926-05	5.6438+03
7	180.00	8.7143-05	3.0285-01	9.6973-03	1.5791-05	1.3452-04	6.7725+03
8	210.00	1.1163-04	3.0038-01	1.1023-02	1.8566-05	1.8491-04	7.9013+03
9	240.00	1.3833-04	3.0004-01	1.2492-02	2.1338-05	2.4293-04	9.0300+03
10	270.00	1.6719-04	3.3367-01	1.3936-02	2.4106-05	3.0844-04	1.0159+04
11	300.00	1.9814-04	3.2743-01	1.5353-02	2.6869-05	3.8128-04	1.1288+04
12	330.00	2.3117-04	3.2126-01	1.6745-02	2.9628-05	4.6130-04	1.2416+04

c)

TIME STEP	TIME DAYS	PU239 GRAMS	U240 GRAMS	PU241 GRAMS	PU242 GRAMS	AM243 GRAMS	BURNUP MWD/TO-U
0	.00	0.0000	0.000	0.0000	0.0000	0.0000	0.0000
1	4.17	0.4092-05	1.661-07	0.0000	0.0000	0.0000	1.5677+02
2	30.00	1.2503-03	1.308-05	5.3030-07	0.0000	0.0000	1.1288+03
3	60.00	2.6021-03	5.2336-05	4.4206-06	0.7207-08	0.0000	2.2575+03
4	90.00	3.8916-03	1.1365-04	1.4551-05	1.4807-07	0.0000	3.3863+03
5	120.00	5.1206-03	1.9314-04	3.3361-05	4.6509-07	0.0000	4.5150+03
6	150.00	6.2918-03	2.8764-04	6.1408-05	1.1019-06	1.0873-08	5.6438+03
7	180.00	7.4081-03	3.9455-04	1.0050-04	2.1906-06	3.0467-08	6.7725+03
8	210.00	8.4724-03	5.1171-04	1.5070-04	3.9025-06	8.3834-08	7.9013+03
9	240.00	9.4870-03	6.3759-04	2.1242-04	6.3602-06	1.8485-07	9.0300+03
10	270.00	1.0454-02	7.7061-04	2.8524-04	8.7472-06	3.2822-07	1.0159+04
11	300.00	1.1377-02	9.0968-04	3.6895-04	1.4100-05	5.4448-07	1.1288+04
12	330.00	1.2256-02	1.0539-03	4.6309-04	1.9842-05	8.5457-07	1.2416+04

1 SYSTEM-MATRIX A

d)

TIME STEP	TIME DAYS	PU251 GRAMS	U252 GRAMS	PU253 GRAMS	U238 GRAMS	NP239 GRAMS	BURNUP MWD/TO-U
0	.00	0.0000	0.000	0.0000	0.8841+00	0.0000	0.0000
1	4.17	0.0000	0.000	0.0000	0.8839+00	1.1324-04	1.5677+02
2	30.00	0.0000	0.0000	0.0000	0.8824+00	1.6224-04	1.1288+03
3	60.00	0.0000	0.0000	0.0000	0.8808+00	1.6398-04	2.2575+03
4	90.00	0.0000	0.0000	0.0000	0.8791+00	1.6543-04	3.3863+03
5	120.00	0.0000	0.0000	0.0000	0.8774+00	1.6672-04	4.5150+03
6	150.00	0.0000	0.0000	0.0000	0.8756+00	1.6798-04	5.6438+03
7	180.00	0.0000	0.0000	0.0000	0.8739+00	1.6925-04	6.7725+03
8	210.00	0.0000	0.000	0.0000	0.8722+00	1.7053-04	7.9013+03
9	240.00	0.0000	0.0000	0.0000	0.8704+00	1.7182-04	9.0300+03
10	270.00	0.0000	0.0000	0.0000	0.8686+00	1.7312-04	1.0159+04
11	300.00	0.0000	0.0000	0.0000	0.8668+00	1.7443-04	1.1288+04
12	330.00	0.0000	0.0000	0.0000	0.8650+00	1.7575-04	1.2416+04

Table(III-2-18):The weights of fuel isotopes for 70%UO_2-30%THO_2 fuel mixture and 16%U-235 enrichment level

a)

TIME STEP	TIME DAYS	K-EFFEKT.	CO YERS. R T*O	TH232 GRAMS	PA233 GRAMS	U233 GRAMS	BURNUP MWD/TO-U
0	.00	1.2051+00	0.0000				
1	4.17	1.1844+00	1.2629-01	1.4004+00	0.0000	0.0000	0.0000
2	30.00	1.1748+00	2.1406-01	1.4003+00	0.0521-05	0.3152-06	1.5678+02
3	60.00	1.1666+00	2.3855-01	1.3997+00	0.3087-04	1.0270-04	1.1288+03
4	90.00	1.1598+00	2.5972-01	1.3991+00	6.3426-04	5.8913-04	2.2576+03
5	120.00	1.1545+00	2.5767-01	1.3984+00	7.3168-04	1.0951-03	3.3864+03
6	150.00	1.1475+00	2.6229-01	1.3978+00	7.7971-04	1.6429-03	4.5152+03
7	180.00	1.1414+00	2.6591-01	1.3971+00	0.0078-04	2.2057-03	5.6440+03
8	210.00	1.1355+00	2.6911-01	1.3964+00	0.1990-04	2.7709-03	6.7728+03
9	240.00	1.1295+00	2.7218-01	1.3958+00	0.2872-04	3.3330-03	7.9016+03
10	270.00	1.1235+00	2.7522-01	1.3951+00	0.3600-04	3.8891-03	9.0304+03
11	300.00	1.1176+00	2.7829-01	1.3944+00	0.4228-04	4.4382-03	1.0159+04
12	330.00	1.1117+00	2.8139-01	1.3937+00	0.4812-04	4.9797-03	1.1288+04
				1.3930+00	0.5379-04	5.5135-03	1.2417+04

b)

TIME STEP	TIME DAYS	U234 GRAMS	U235 GRAMS	U236 GRAMS	U237 GRAMS	NP237 GRAMS	BURNUP MWD/TO-U
0	.00	0.0000	5.4734-01	0.0000	0.0000	0.0000	0.0000
1	4.17	6.5086-09	5.2335-01	0.4669-04	6.5723-08	0.0000	1.5678+02
2	30.00	2.9644-06	5.1735-01	1.7756-03	1.7930-06	1.6779-06	1.1288+03
3	60.00	1.0280-05	5.1021-01	3.5243-03	4.3492-06	1.1042-05	2.2576+03
4	90.00	2.0524-05	5.0335-01	5.2734-03	6.9632-06	2.0242-05	3.3864+03
5	120.00	3.3013-05	4.9630-01	6.9816-03	9.5805-06	3.3210-05	4.5152+03
6	150.00	4.7414-05	4.8941-01	8.6884-03	1.2128-05	4.5864-05	5.6440+03
7	180.00	6.3560-05	4.8259-01	1.0364-02	1.4817-05	1.2613-04	6.7728+03
8	210.00	8.1364-05	4.7591-01	1.2019-02	1.7436-05	1.7584-04	7.9016+03
9	240.00	1.0077-04	4.6934-01	1.3651-02	2.0057-05	2.2890-04	9.0304+03
10	270.00	1.2176-04	4.6282-01	1.5262-02	2.2620-05	2.8121-04	1.0159+04
11	300.00	1.4428-04	4.5650-01	1.6853-02	2.5305-05	3.6072-04	1.1288+04
12	330.00	1.6834-04	4.4993-01	1.8422-02	2.7931-05	4.3726-04	1.2417+04

c)

TIME STEP	TIME DAYS	PU251 GRAMS	PU252 GRAMS	PU253 GRAMS	U238 GRAMS	NP239 GRAMS	BURNUP MWD/TO-U
0	.00	0.0000	0.000	0.0000	0.7528+00	0.0000	0.0000
1	4.17	0.0000	0.000	0.0000	0.7527+00	0.5271-05	1.5678+02
2	30.00	0.0000	0.000	0.0000	0.7514+00	1.3608-04	1.1288+03
3	60.00	0.0000	0.000	0.0000	0.7500+00	1.3730-04	2.2576+03
4	90.00	0.0000	0.000	0.0000	0.7486+00	1.3839-04	3.3864+03
5	120.00	0.0000	0.000	0.0000	0.7471+00	1.3938-04	4.5152+03
6	150.00	0.0000	0.000	0.0000	0.7457+00	1.4025-04	5.6440+03
7	180.00	0.0000	0.000	0.0000	0.7442+00	1.4114-04	6.7728+03
8	210.00	0.0000	0.000	0.0000	0.7428+00	1.4203-04	7.9016+03
9	240.00	0.0000	0.000	0.0000	0.7413+00	1.4293-04	9.0304+03
10	270.00	0.0000	0.000	0.0000	0.7398+00	1.4384-04	1.0159+04
11	300.00	0.0000	0.000	0.0000	0.7383+00	1.4475-04	1.1288+04
12	330.00	0.0000	0.000	0.0000	0.7368+00	1.4568-04	1.2417+04

d)

TIME STEP	TIME DAYS	U239 GRAMS	U240 GRAMS	PU241 GRAMS	PU242 GRAMS	AM243 GRAMS	BURNUP MWD/TO-U
0	.00	0.0000	0.000	0.0000	0.0000	0.0000	0.0000
1	4.17	7.0803-05	1.904-07	0.0000	0.0000	0.0000	1.5678+02
2	30.00	1.0562-03	8.359-06	2.7674-07	0.0000	0.0000	1.1288+03
3	60.00	2.2117-03	3.465-05	2.3194-06	0.0000	0.0000	2.2576+03
4	90.00	3.3290-03	7.3302-05	7.7230-06	4.5746-08	0.0000	3.3864+03
5	120.00	4.4083-03	1.2607-04	1.7785-05	1.7005-07	0.0000	4.5152+03
6	150.00	5.4504-03	1.8976-04	3.3493-05	4.2055-07	0.0000	5.6440+03
7	180.00	6.4564-03	2.6294-04	5.5615-05	8.5554-07	0.0000	6.7728+03
8	210.00	7.4275-03	3.4435-04	8.4660-05	1.5041-06	0.0000	7.9016+03
9	240.00	8.3649-03	4.3288-04	1.2102-04	2.5666-06	2.4163-08	9.0304+03
10	270.00	9.2699-03	5.2760-04	1.6985-04	3.9998-06	6.2824-08	1.0159+04
11	300.00	1.0143-02	6.2773-04	2.1621-04	5.8844-06	1.3423-07	1.1288+04
12	330.00	1.0986-02	7.3259-04	2.7509-04	8.3307-06	2.3680-07	1.2417+04

SYSTEM-MATRIX A

Table(III-2-19):The weights of fuel isotopes for 70%UO_2-30%THO_2 fuel mixture and 20%U-235 enrichment level

a)

TIME STEP	TIME DAYS	K-EFFEKT.	CO VERS. RATIO	TH232 GRAMS	PA233 GRAMS	U233 GRAMS	BURNUP MWD/TO-U
0	.00	1.2456+00	0.0000	1.4004+00	0.0000	0.0000	0.0000
1	4.17	1.2283+00	1.0738-01	1.4003+00	7.2075-05	3.8628-06	1.5678+02
2	30.00	1.2200+00	1.8340-01	1.3998+00	3.8523-04	1.6354-04	1.1288+03
3	60.00	1.2124+00	2.0389-01	1.3992+00	5.6684-04	5.2790-04	2.2576+03
4	90.00	1.2060+00	2.1433-01	1.3986+00	6.5368-04	9.8244-04	3.3864+03
5	120.00	1.2001+00	2.2013-01	1.3980+00	6.9634-04	1.4757-03	4.5152+03
6	150.00	1.1946+00	2.2382-01	1.3974+00	7.1840-04	1.9838-03	5.6440+03
7	180.00	1.1893+00	2.2657-01	1.3968+00	7.3088-04	2.4955-03	6.7728+03
8	210.00	1.1839+00	2.2892-01	1.3962+00	7.3890-04	3.0057-03	7.9016+03
9	240.00	1.1787+00	2.3113-01	1.3956+00	7.4486-04	3.5119-03	9.0304+03
10	270.00	1.1734+00	2.3330-01	1.3950+00	7.4988-04	4.0130-03	1.0159+04
11	300.00	1.1682+00	2.3548-01	1.3944+00	7.5448-04	4.5085-03	1.1288+04
12	330.00	1.1630+00	2.3770-01	1.3938+00	7.5892-04	4.9982-03	1.2417+04

b)

TIME STEP	TIME DAYS	U234 GRAMS	U2 5 GR MS	U236 GRAMS	U237 GRAMS	NP237 GRAMS	BURNUP MWD/TO-U
0	.00	0.0000	6.5542-01	0.0000	0.0000	0.0000	0.0000
1	4.17	5.1409-08	6.5442-01	2.6077-04	6.1875-08	0.0000	1.5678+02
2	30.00	2.3236-06	6.4821-01	1.8773-03	1.6734-06	2.0613-06	1.1288+03
3	60.00	8.0397-06	6.4104-01	3.7409-03	4.0799-06	1.0728-05	2.2576+03
4	90.00	1.6029-05	6.3390-01	5.5885-03	6.5342-06	2.6979-05	3.3864+03
5	120.00	2.5755-05	6.2680-01	7.4190-03	8.9960-06	5.0514-05	4.5152+03
6	150.00	3.6959-05	6.1974-01	9.2320-03	1.1461-05	8.1308-05	5.6440+03
7	180.00	4.9513-05	6.1271-01	1.1027-02	1.3929-05	1.1943-04	6.7728+03
8	210.00	6.3350-05	6.0573-01	1.2804-02	1.6400-05	1.6456-04	7.9016+03
9	240.00	7.8434-05	5.9878-01	1.4563-02	1.8874-05	2.1661-04	9.0304+03
10	270.00	9.4739-05	5.9187-01	1.6304-02	2.1352-05	2.7578-04	1.0159+04
11	300.00	1.1225-04	5.8500-01	1.8028-02	2.3833-05	3.4183-04	1.1288+04
12	330.00	1.3095-04	5.7817-01	1.9733-02	2.6318-05	4.1468-04	1.2417+04

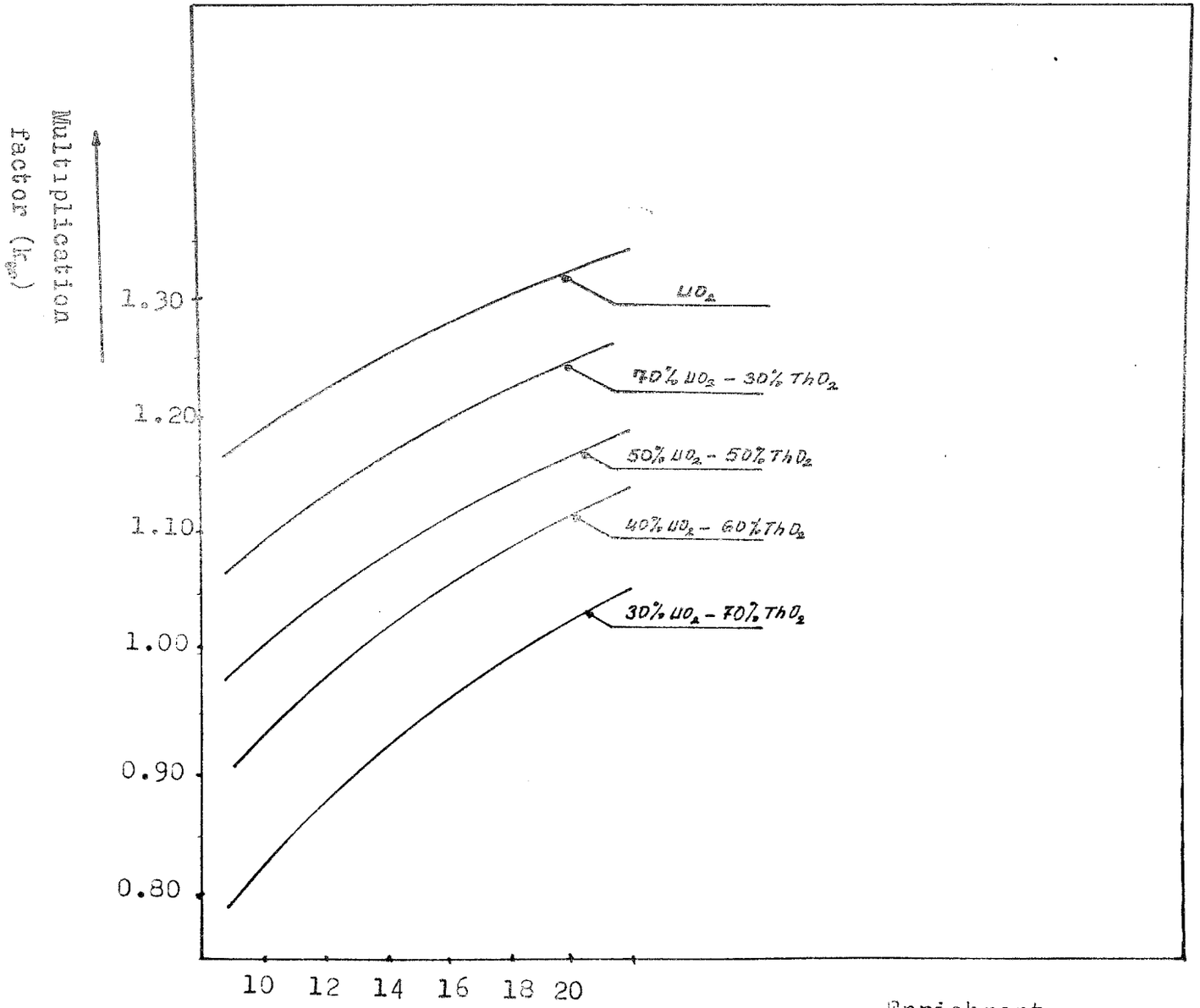
c)

TIME STEP	TIME DAYS	POIS1 GRAMS	P IS2 GR MS	POIS3 GRAMS	U238 GRAMS	NP239 GRAMS	BUNUP MWD/TO-U
0	.00	0.0000	0. 000	0.0000	2.6218+00	0.0000	0.0000
1	4.17	0.0000	0.0000	0.0000	2.6216+00	8.2137-05	1.5678+02
2	30.00	0.0000	0.0000	0.0000	2.6205+00	1.1700-04	1.1288+03
3	60.00	0.0000	0.0000	0.0000	2.6193+00	1.1789-04	2.2576+03
4	90.00	0.0000	0.0000	0.0000	2.6181+00	1.1873-04	3.3864+03
5	120.00	0.0000	0.0000	0.0000	2.6168+00	1.1947-04	4.5152+03
6	150.00	0.0000	0.0000	0.0000	2.6156+00	1.2017-04	5.6440+03
7	180.00	0.0000	0.0000	0.0000	2.6143+00	1.2084-04	6.7728+03
8	210.00	0.0000	0.0000	0.0000	2.6130+00	1.2151-04	7.9016+03
9	240.00	0.0000	0.0000	0.0000	2.6118+00	1.2217-04	9.0304+03
10	270.00	0.0000	0.0000	0.0000	2.6105+00	1.2284-04	1.0159+04
11	300.00	0.0000	0.0000	0.0000	2.6092+00	1.2351-04	1.1288+04
12	330.00	0.0000	0.0000	0.0000	2.6079+00	1.2419-04	1.2417+04

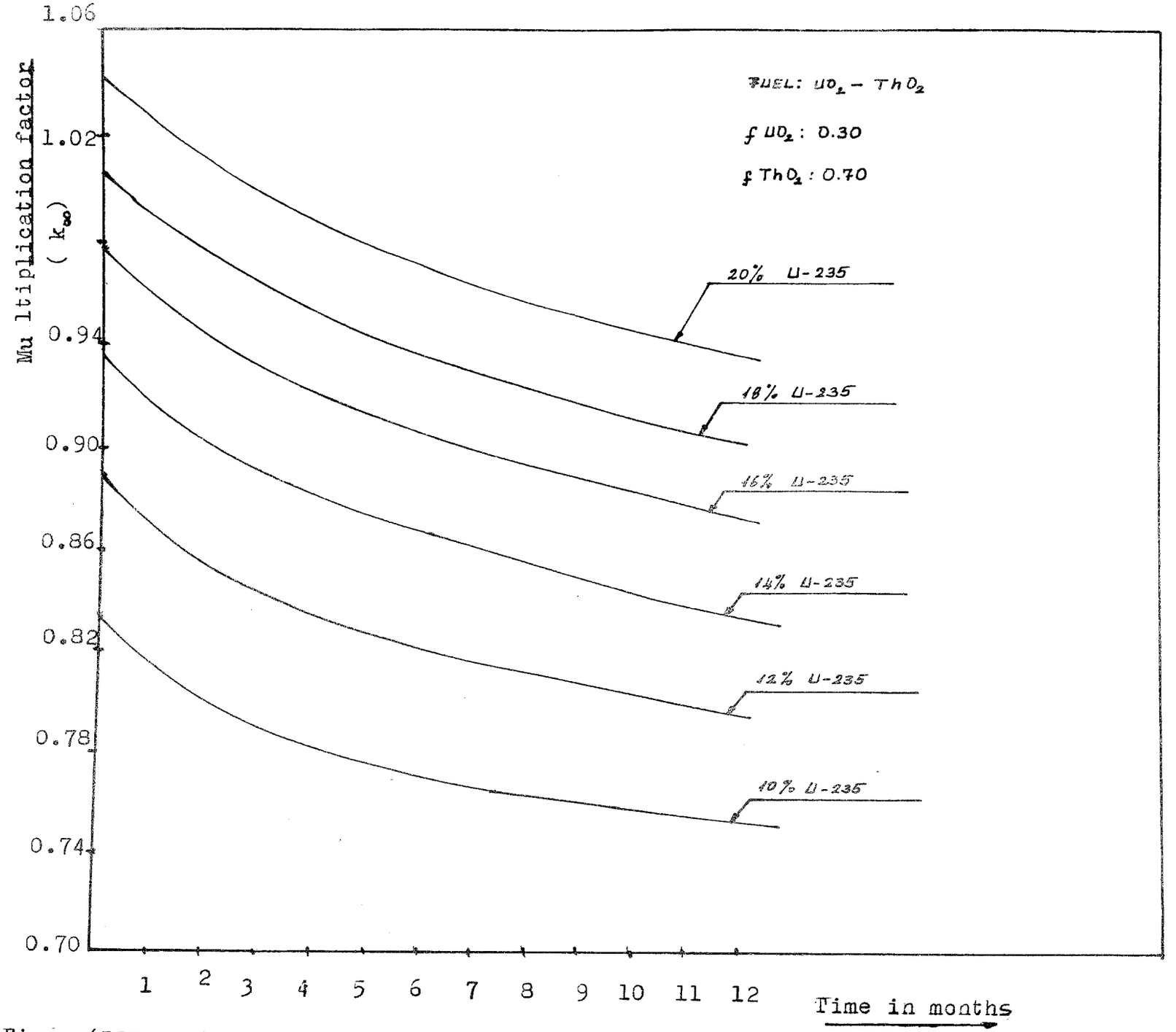
d)

TIME STEP	TIME DAYS	PU239 GRAMS	U240 RAMS	PU241 GRAMS	PU242 GRAMS	AM243 GRAMS	BURNUP MWD/TO-U
0	.00	0.0000	0. 000	0.0000	0.0000	0.0000	0.0000
1	4.17	6.1069-05	8. 860-08	0.0000	0.0000	0.0000	1.5678+02
2	30.00	9.1229-04	5.6816-06	3.8364-08	0.0000	0.0000	1.1288+03
3	60.00	1.9174-03	2.2932-05	1.2149-06	0.0000	0.0000	2.2576+03
4	90.00	2.8973-03	5.0588-05	4.3607-06	0.0000	0.0000	3.3864+03
5	120.00	3.8519-03	8.7512-05	1.0262-05	1.1236-08	0.0000	4.5152+03
6	150.00	4.7813-03	1.3268-04	1.9617-05	1.2119-07	0.0000	5.6440+03
7	180.00	5.6859-03	1.8517-04	3.2977-05	3.2504-07	0.0000	6.7728+03
8	210.00	6.5663-03	2.4418-04	5.0789-05	6.3563-07	0.0000	7.9016+03
9	240.00	7.4229-03	3.0899-04	7.3351-05	1.1108-06	0.0000	9.0304+03
10	270.00	8.2563-03	3.7899-04	1.0093-04	1.7737-06	0.0000	1.0159+04
11	300.00	9.0670-03	4.5360-04	1.3367-04	2.6774-06	0.0000	1.1288+04
12	330.00	9.8557-03	5.3236-04	1.7171-04	3.8580-06	0.0000	1.2417+04

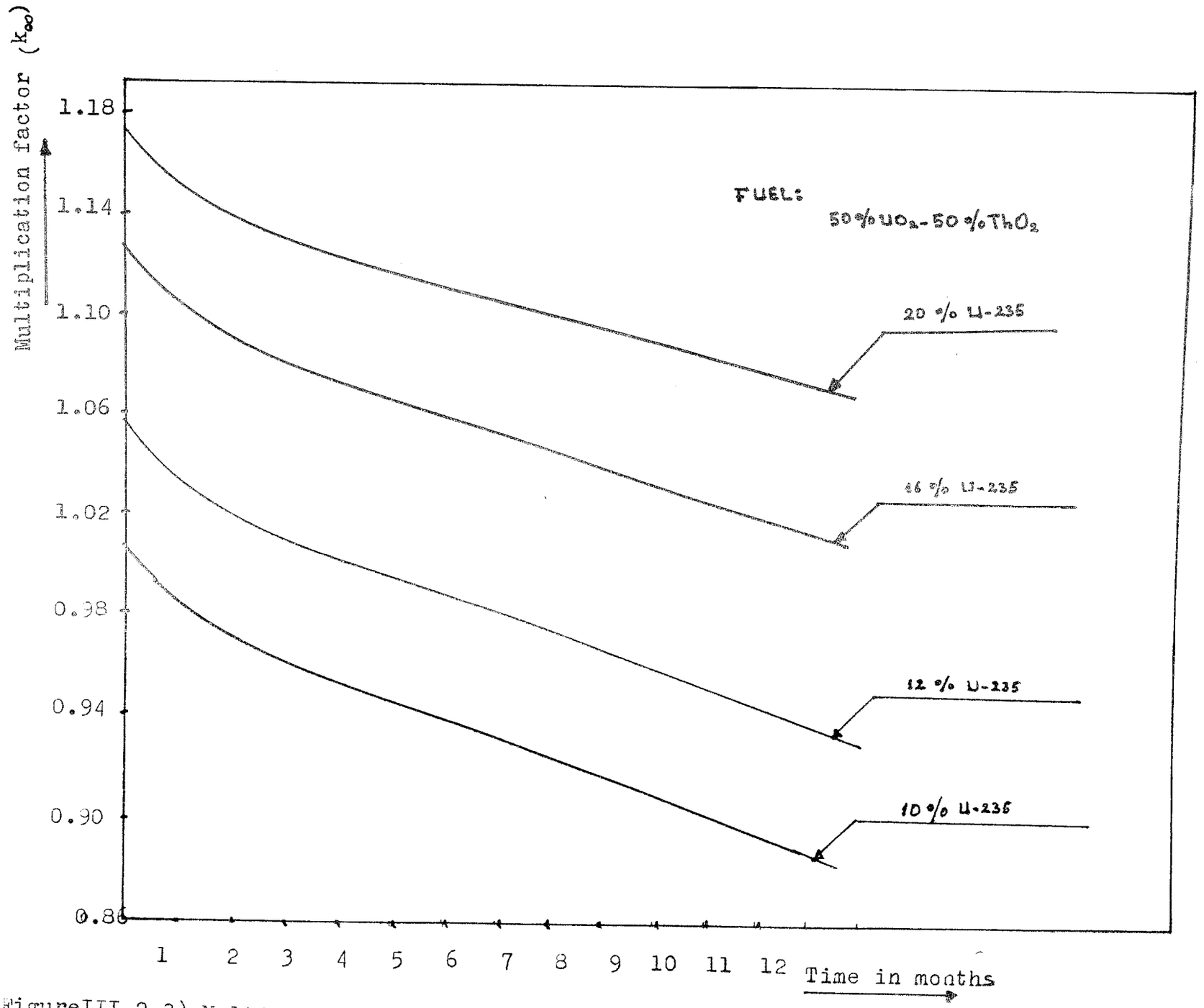
1 SYSTEM-MATRIX A



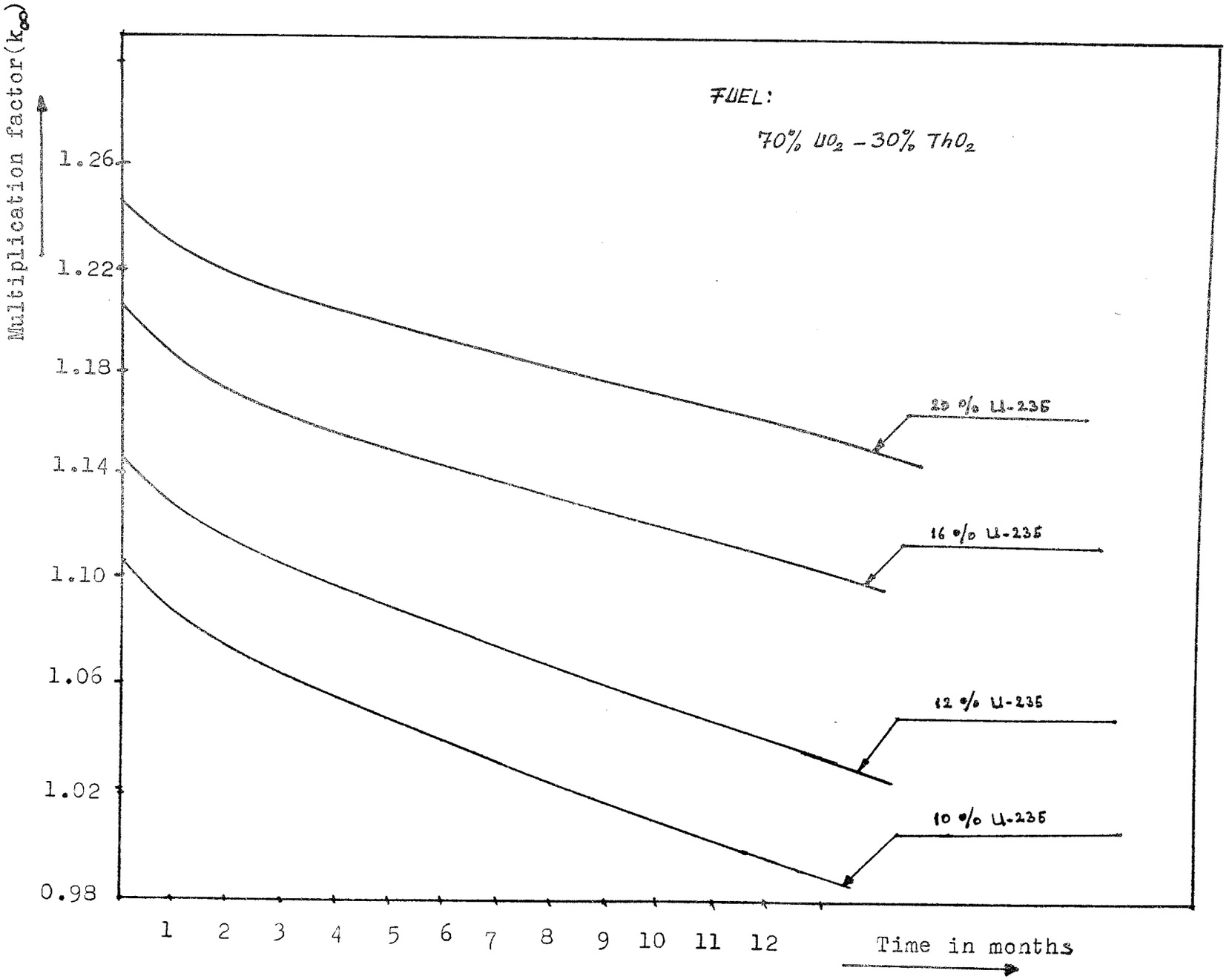
Figure(III-2-1):Multiplication factor versus enrichment for different mixtures



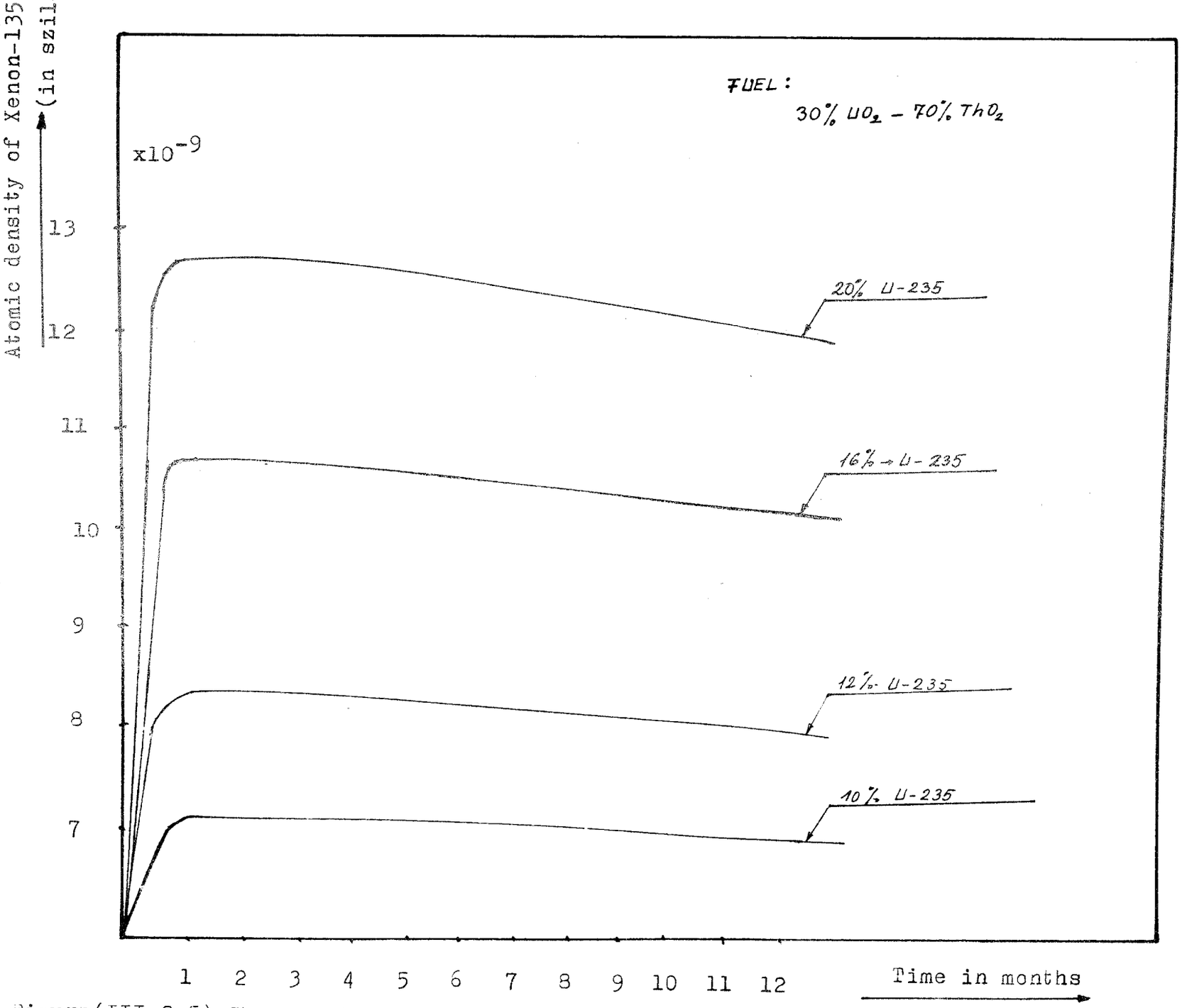
Figure(III-2-2):Multiplication factor versus time



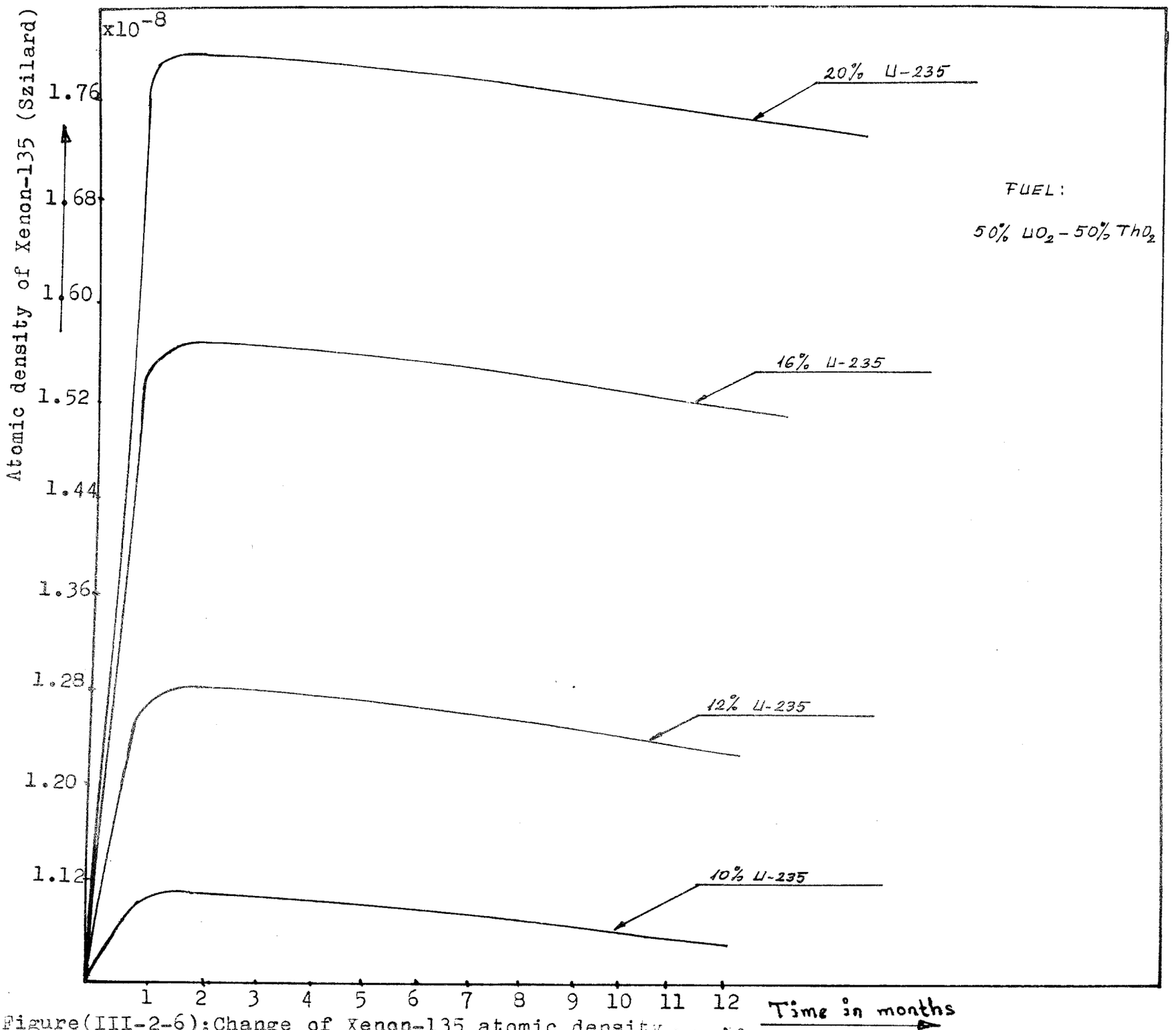
FigureIII-2-3):Multiplication factor versus time



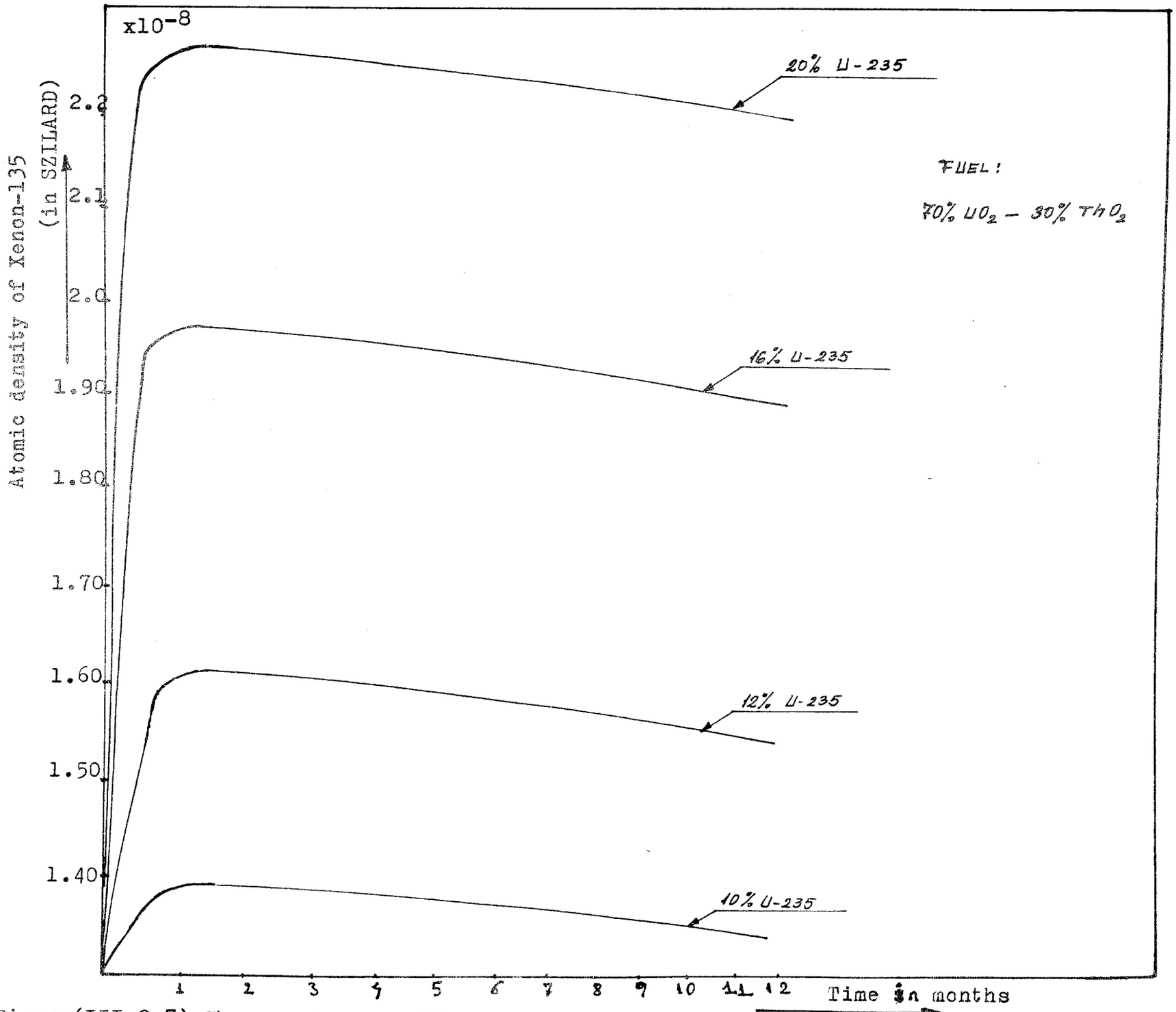
Figure(III-2-4):Multiplication factor versus time



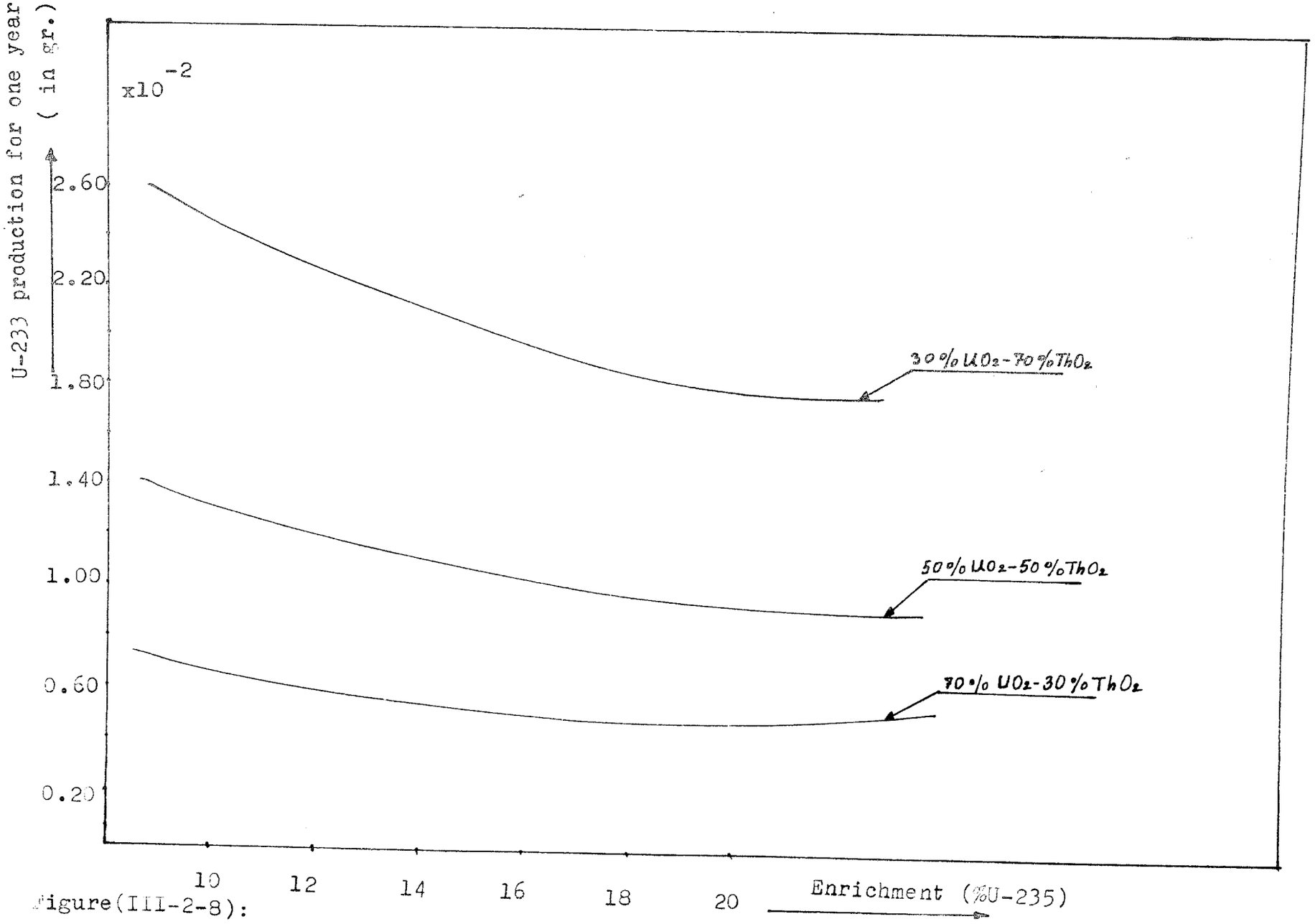
Figure(III-2-5):Change of Xenon-135 atomic density with time



Figure(III-2-6):Change of Xenon-135 atomic density with time



Figure(III-2-7):Change of Xenon-135 atomic density with time



Figure(III-2-8): U-233 buildup for different UO_2-THO_2 mixtures and U-235 enrichment levels

APPENDIX A

INPUT DESCRIPTION

A. Card Description

Card No.

1	(20A4)	Title card
2	(18I4)	Basic control and dimensioning parameters (See the discussion in section C for definitions of "region", "zone", and various types of "cells".)
col.1 - 4	NBURP	Lumped absorber specification
	= 0	No lumped absorber
	= 1	Gd 155, Gd 156, Gd 157
	= 2	Silver, Cd 113, In 115
	= 3	Cd 113
	= 4	B 10
col. 5 - 8	N2O	Number of regions, sum total for all zones ($N2O \geq 3$). Input of $N2O < 3$ causes termination of the run
col. 9 - 12	NGR	Number of condensed (broad) groups
col.13 - 16	NZM	Number of zones with different moderator material (zones without moderator neglected)

Card No.

2 (cont.) i.e. number of moderators. Only $H - \frac{1}{2} O$ is in library. Only different moderator densities are meant here.

col.17 - 20 ISTO — Case type declaration (See Chapter 2, Section 2.3.1)

= 1 Normal cell calculation, coefficients of the self shielding polynomial of the fuel and related isotopes are generated and written to logical I/O device 2. See also Section D.

= 2 Burn-up supercell calculation, the coefficients generated by ISTO = 1 have to be used.

= 3 Normal cell calculation, full output possible.

= 4 Normal cell calculation, only homogenized library will be generated on logical file NUY. (NBURP > 0)

= 5 Control supercell calculation, the homogenized library generated by ISTO = 4 has to be used.

3 (6E12.5) Input of temperatures

col. 1-12 TBRENN Fuel temperature ^{o}K

col.13-24 TMOD Moderator temperature ^{o}K

Card No.

4	(18I4)		Output control parameters
	col. 1 - 4	NPRIN3	Print index of the flux field
		= 0	45-group fluxes are printed
		= 1	2-group fluxes are printed (fast and thermal)
	col. 5 - 8	NPRIN5	Print index of the nuclid dependent correction factors (EREBUS)
		= 0	No print
		= 1	Print
	col. 9 - 12	NPUNC	Output index of the condensed cross section library
		= 1	Write EREBUS input to mass storage (See also Card 16)
		= 0	No punch
	col. 13 - 16	NPRIN1	
		= 0	No effect
		= 1	Suppression of all except synopsis output
5	(20A4)		Subtitle card for further specification
6	(18I4)		Case control numbers
	col. 1 - 4	NXXX	Punch index for region dependent densities (cards No. 12, 13) for restart from a certain time step
		= 0	No punch
		= 1	Punch

Card No.

6 (cont.)

col. 5 - 8	NRSTRT	Time step number of density punch. No effect if NXXX = 0
col. 9 - 12	NDOW	Group condensation control number
	= 0	No group condensation performed
	= 1	Group condensation performed
col.13 - 16	KSER	Initiation of successor cases
	= 0	No new case
	= 1	Full new case starting witn card 1
	= -1	New case starting with card 5
col.17 - 20	JAN	Control for special sequential cases. No effect if KSER = 0 in the preceding case.
	= 0	No effect
	= 1	Repeated homogenization with change in regions involved. (cf. card No. 7) The new input consists only of cards No. 5, 6, 7). KSER = -1 in the preceding case.
	= 2	Repetition of a case with moderator densities altered. The new input consist of cards No. 1-6 and card No.8. KSER = 1 in the preceding case.

7 (18I4) Regional boundaries for homogenization

Supplied only if JAN = 1 or ISTO = 5

Card No.

7 (cont.)

col. 1 - 4	IGC1	First	region participating in
col. 5 - 8	IGC2	Last	homogenization

8 (6E12.5) Moderator densities

col. 1 - 12	DENM(1)	Density in the first zone containing moderator
	.	
	.	
	DENM(NZM)	Density in the last zone containing moderator

9 (18I4) Boundaries of condensed groups

(omitted if NDOW = 0)

col. 1 - 4	ICPU(1)	Number of the highest small group,	within broad group 1
col. 5 - 8	ICP(1)	Number of the lowest small group	
	.		
	.		
	.		
	ICPU(NGR)	Number of the highest small group	within broad group NGR
	ICP(NGR)	Number of the lowest small group	

Note: The group of the highest energy has the lowest number.
The option ISTO = 4 demands for the input of card 9 (and for NDOW ≠ 0)
Else no library can be produced.

Card No.			
10-11			Geometry and related input
10 (18I4)			Specification of zones, attachment of regions and moderators to zones.
	col. 1 - 4	NZO	Number of zones
	col. 5 - 8	MP(1)	Number of regions in zone 1
	col. 9 - 12	IT(1)	Moderator index of zone 1

- -

MP(NZO)

IT(NZO)

The regions of a zone are annuli of equal thickness

IT(J) = K: Zone J contains moderator K.

Moderators are indexed by the sequence of their densities (cf. card No. 8).

$1 \leq K \leq NZM$

IT(J) = -1: No moderator in zone J.

11 (6E12.5)			Specification of radii
	col. 1 - 12	RZ(1)	Outer radius of zone 1
	col.13 - 24	RZ(2)	Outer radius of zone 2
		.	
		.	
		RZ(NZO)	Outer radius of zone NZO
12-13			Region dependent atomic densities
			This set of cards has to be supplied for every region IR; $1 \leq IR \leq NZO$

Card No.

12	(18I4)		Control card
	col. 1 - 4	NREAD	Number of cards No. 13 having to follow card No. 12
	col. 5 - 8	NCH1	Previous region specifier
		= 0	If NREAD ≠ 0
		≠ 0	If NREAD = 0
			Designates a foregoing region with identical atomic densities.
13	(I4,8X,E12.5)		Supplied only for NREAD > 0
	col. 1 - 4	L	Isotope number (fixed by library, see Section B, following)
	col.13 - 24	DEN(L,IR)	Atomic density of isotope L ($barn^{-1} * cm^{-1}$)
14	(6E12.5)		Liquid boron content of the moderator and burn-up data
	col. 1 - 12	PPM	Parts per million of natural boron
	col.13 - 24	POWER	Integral power per cm in Watts
	col.25 - 36	STOP	Number of burn-up time steps
	col.37 - 48	DELDAY	Burn-up time step in days
	col.49 - 60	ZKFIND	Minimum k_∞ terminating the burn-up calculation.

Notes: POWER ≠ .0, or else print of the flux field is wrong.
STOP = .0, must be put in if no burn-up is wanted.

Card No.

15 (3I4, E12.5) Special input "Burn-up supercell" data

This card has to be omitted if ISTO ≠ 2.

col. 1 - 4 ICON Isotope number of the "leading density" (must be 5 for present version of code)

col. 5 - 8 IGC1 First } region participating

col. 9 - 12 IGC2 Last } in homogenization

Note: The remaining data for the burn-up supercell is read from logical I/O device 2.

16 (2I4) Control card for writing output that will be used for EREBUS input.

col. 1 - 4 IEREB Mass storage logical I/O unit number to which input for EREBUS is to be written. Should be 23 for first set of data to be read by EREBUS, 24 for recond set, etc. See also Section D.

NPUNC

= 0 Do not write EREBUS input

= 1 Write EREBUS input

(One card number 17 must be supplied for each time step - including the zeroth - up to the last time step for which it is desired to write EREBUS input. The value of IEREB must be different for each time step at which EREBUS input is to be

APPENDIX B

SERIES OF ISOTOPES IN THE LIBRARY

I. Heavy Metals (FUEL = 15)

1	Thorium	232
2	Protoactinium	233
3	Uranium	233
4	Uranium	234
5	Uranium	235
6	Uranium	236
7	Uranium	237
8	Neptunium	237
9	Uranium	238
10	Neptunium	239
11	Plutonium	239
12	Plutonium	240
13	Plutonium	241
14	Plutonium	242
15	Americium	243

II. Fission Products (NLT = 31)

16	Non saturating fission product of Th^{232}		
17	Non saturating fission product of U^{233}		
18	Non saturating fission product of U^{235}		
19	Non saturating fission product of U^{238}		
20	Non saturating fission product of Pu^{239}		
21	Non saturating fission product of Pu^{241}		
22	Rhodium	103	
23	Iodine	131	First chain
24	Xenon	131	
25	Xenon	133	Second chain
26	Cesium	133	
27	Cesium	134	
28	Iodine	135	
29	Xenon	135	
30	Cesium	135	
31	Praseodymium	143	Third chain
32	Neodymium	143	
33	Molybdenum	95	
34	Technetium	99	
35	Neodymium	145	
36	Neodymium	146	
37	Neodymium	147	Fourth chain
38	Promethium	147	

39	Promethium	148
40	Promethium	148
41	Promethium	149
42	Samarium	149
43	Samarium	150
44	Promethium	151
45	Samarium	151
46	Samarium	152

III. Lumped absorbers

47	Gadolinium	155	NBuRP = 1
48	Gadolinium	156	
49	Gadolinium	157	
47	Silver	109	NBuRP = 2
48	Cadmium	113	
49	Indium	115	
47	Cadmium	113	NBuRP = 3
47	Boron	10	NBuRP = 4

IV. Nonburnable isotopes (NOB = 12)

47	Boron	10
48	Chromium	
49	Zirconium	
50	Oxygen (in UO_2, see also V. for O in H_2O)	
51	Nickel	

52 Iron

53 Titanium

54 Manganese

55 Copper

56 Niobium

57 Molybdenum

58 Aluminum

APPENDIX C

SAMPLE INPUT FOR FUEL CELL CALCULATION

Zone	Radius(in cm)
1	0,418
2	0,475
3	0,6991
4	0,7296

Number Densities

Zone	Element	Number density(SZILARD)
1	U-235 U-238 TH-232 O_2	See tables given below
2	Cr	$7,6012164.10^{-5}$
	Fe	$1,5547943.10^{-4}$
	Ni	$4,7507603.10^{-6}$
	Zr	$4,2454089.10^{-2}$
3	Cr	$7,1273560.10^{-4}$
	Mn	$1,768237.10^{-5}$
	Fe	$8,545346.10^{-4}$
	Ni	$1,745454.10^{-4}$
	Nb	$1,101774.10^{-3}$
	Cu	$2,312310.10^{-6}$
	Ti	$3,230433.10^{-6}$
	Mo	$6,222834.10^{-6}$
	HoL/2	$4,569669.10^{-2}$
4	Cr	$2,410121.10^{-3}$
	Mn	$1,829595.10^{-3}$
	Fe	$7,185106.10^{-3}$
	Ni	$2,353043.10^{-3}$
	Nb	$8,15627.10^{-3}$
	Cu	$1,701181.10^{-6}$
	Ti	$2,376601.10^{-6}$
	Mo	$4,578178.10^{-6}$
	HO1/2	$4,088135.10^{-2}$

TABLE C-1

Number densities for 30%UO_2-70%THO_2 fuel mixture

Enrichment %U-235	N_{235}	N_{238}	NO_2	N_{Th}
10	$6,5563410^{-4}$	$5,82633.10^{-3}$	$2,19460.10^{-2}$	$1,59707.10^{-2}$
12	$7,8679.10^{-4}$	$5,69706.10^{-3}$	$2,19473.10^{-2}$	"
14	$9,17894.10^{-4}$	$5,56742.10^{-3}$	$2,19487.10^{-2}$	"
16	$1,04899.10^{-3}$	$5,43777.10^{-3}$	$2,19502.10^{-2}$	"
18	$1,18007.10^{-3}$	$5,30810.10^{-3}$	$2,19518.10^{-2}$	"
20	$1,31115.10^{-3}$	$5,17852.10^{-3}$	$2,19530.10^{-2}$	"

TABLE C-2

Number densities for40%UO_2-60%THO_2 fuel mixture

Enrichment % U-235	N_{235}	N_{238}	N_{02}	N_{Th}
10	$8,74260.10^{-4}$	$7,76915.10^{-3}$	$2,18978.10^{-2}$	$1,32435.10^{-2}$
12	$1,04907.10^{-3}$	$7,59626.10^{-3}$	$2,19029.10^{-2}$	"
14	$1,22389.10^{-3}$	$7,42342.10^{-3}$	$2,19016.10^{-2}$	"
16	$1,39869.10^{-3}$	$7,25057.10^{-3}$	$2,19035.10^{-2}$	"
18	$1,57347.10^{-3}$	$7,07769.10^{-3}$	$2,19057.10^{-2}$	"
20	$1,7426.10^{-3}$	$6,90489.10^{-3}$	$2,19073.10^{-2}$	"

TABLE C-3

Number densities for 50%UO_2-50%THO_2 fuel mixture

Enrichment %U-235	N_{235}	N_{238}	N_{02}	N_{Th}
10	$1,09094.10^{-3}$	$9,699170.10^{-3}$	$2,1803.10^{-2}$	$1,006.10^{-2}$
12	$1,30927.10^{-3}$	$9,479040.10^{-3}$	$2,18057.10^{-2}$	"
14	$1,52720.10^{-3}$	$9,26192.10^{-3}$	$2,18203.10^{-2}$	"
16	$1,74536.10^{-3}$	$9,04653.10^{-3}$	$2,18103.10^{-2}$	"
18	$1,96346.10^{-3}$	$8,83232.10^{-3}$	$2,18130.10^{-2}$	"
20	$2,18157.10^{-3}$	$8,61631,10^{-3}$	$2,18150.10^{-2}$	"

TABLE C-4

Number densities for 70%UO_2-30%THO_2 fuel mixture

Enrichment % U-235	N_{235}	N_{238}	N_{02}	N_{Th}
10	$1,52997.10^{-3}$	$1,35960.10^{-2}$	$2,17531.10^{-2}$	$6,621178.10^{-3}$
12	$1,83590.10^{-3}$	$1,32935.10^{-2}$	$2,17559.10^{-2}$	"
14	$2,14181.10^{-3}$	$1,29909.10^{-2}$	$2,17598.10^{-2}$	"
16	$2,44771.10^{-3}$	$1,26885.10^{-2}$	$2,17633.10^{-2}$	"
18	$2,75358.10^{-3}$	$1,23859.10^{-2}$	$2,17670.10^{-2}$	"
20	$3,05965.10^{-3}$	$1,20843.10^{-2}$	$2,17725.10^{-2}$	"

TABLE C-5

Number densities for 90%UO_2-10%THO_2 fuel mixture

Enrichment %U-235	N_{235}	N_{238}	N_{O2}	N_{Th}
10	1,96708.10^{-3}	1,74806.10^{-2}	2,16566.10^{-2}	2,20720.10^{-3}
12	2,36040.10^{-3}	1,70916.10^{-2}	2,16613.10^{-2}	"
14	2,75340.10^{-3}	1,67010,10^{-2}	2,16800.10^{-2}	"
16	3,14700.10^{-3}	1,63137.10^{-2}	2,16695.10^{-2}	"
18	3,54030.10^{-3}	1,59248.10^{-2}	2,16745.10^{-2}	"
20	3,93350.10^{-3}	1,55360.10^{-2}	2,16781.10^{-2}	"

TABLE C-6

Number densities for pure UO_2 fuel

Enrichment % U-235	N_{235}	N_{238}	N_{O2}
10	2,18565.10^{-3}	1,94229.10^{-2}	2,16084.10^{-2}
12	2,62229.10^{-3}	1,89906.10^{-2}	2,16136.10^{-2}
14	3,05972.10^{-3}	1,85580.10^{-2}	2,16180.10^{-2}
16	3,49673.10^{-3}	1,81264.10^{-2}	2,16227.10^{-2}
18	3,93368.10^{-3}	1,76694.10^{-2}	2,16282.10^{-2}
20	4,37706.10^{-3}	1,72622.10^{-2}	2,16323.10^{-2}

APPENDIX D

SAMPLE RUN

1-LISTING OF INPUT DATA FOR NORMAL FUEL CELL CALCULATION (ISTO =3)

```
@HDG    KWO CLEAN ZERO POWER
@ASG,T     GERBTST.
@USE     24,GERBTST.
@ASG,T     2.
@ASG,T    12.
@ASG,T    13.
@ASG,T    14.
@ASG,T    15.
@ASG,T    16.
@ASG,T    21.
@ASG,T    22.
@ASG,AX    BUNE*GELSLIB.
@USE    17,BUNE*GELSLIB.
@ASG,AX    BUNE*ABS.
@XQT       BUNE*ABS.GELS
 KWO CLEAN ZERO POWER
   0    5    7    2    3
   553.00000    553.00000
   1    0    0
 KWO ENRICHMENT    2.58
   0    0    1    0    0
       .05047         .04658
   1   15   16   45
   4    2   -1    1   -1    1    1    1    2
       .46500         .53700         .81000         .84525
   3    0
   5  U5  2.5   5.54650-004
   9  U8        2.13580-002
  50  OX        4.38250-002
   0
   4    0
  48  CR        7.60120-005
  52  FE        1.55480-004
  51  NI        4.75070-006
  49  ZR        4.24540-002
   8    0
  48  CR        2.02690-004
  51  NI        4.96380-004
  52  FE        2.43010-004
  53  TI        9.18680-007
  54  MN        5.02850-006
  55  CU        6.57560-007
  56  NB        3.13320-006
  57  MO        1.76970-006
   8    0
  48  CR        1.48530-003
  51  NI        1.05570-003
  52  FE        4.90540-003
  53  TI        5.14910-007
  54  MN        1.26440-004
  55  CU        3.68570-007
  56  NB        1.75610-006
  57  MO        9.91890-007
  2000.          151.          0.          50.          .89
  24
```

2-LISTING OF INPUT DATA FOR BURNUP FUEL CELL CALCULATION (ISTO=1)

```
1        ΔASG,A    GE_SCOEFZ.
2        ΔJSE    2,GE_SCOEFZ.
3        ΔASG,T    JGJRFILETWO.,F24
4        ΔJSE   13,JGJRFILETWO.
5        ΔASG,T    JGJRFILETHR.,F24
6        ΔJSE   12,JGJRFILETHR.
7        ΔASG,T    JGJRFILEFOJ.,F24
8        ΔJSE   14,JGJRFILEFOJ.
9        ΔASG,T    JGJRFILEFIV.,F24
10       ΔJSE   22,JGJRFILEFIV.
11       ΔASG,T    JGJRFILESIX.,F24
12       ΔJSE   15,JGJRFILESIX.
13       ΔASG,T    JGJRFILESEV.,F24
14       ΔJSE   16,JGJRFILESEV.
15       ΔASG,T    JGJRFILEEIG.,F24
16       ΔJSE   21,JGJRFILEEIG.
17       ΔASG,A    BJTEK*GE_SDATA.
18       ΔJSE   17,BJTEK*GE_SDATA.
19       ΔXQT    CSEC.MAIN
20        KWO CLEAN ZERO POWER
21          0    5    2    2    1
22          553.00000    553.00000
23          1    1    0
24        KWO ENRICHMENT    2.5%
25          0    0    1    0    0
26              .05047         .04058
27          1   15   16   45
28          4    2   -1    1   -1    1    1    1    2
29              .46500         .53700         .81000         .84525
30          3    0
31          5 U5 2.5  5.54650-004
32          9 U8      2.13580-002
33         50 OX      4.38250-002
34          0    1
35          4    0
36         48 CR      7.60120-005
37         52 FE      1.55480-004
38         51 NI      4.75070-006
39         49 ZR      4.24540-002
40          8    0
41         48 CR      2.02690-004
42         51 NI      4.96380-004
43         52 FE      2.43010-004
44         53 TI      9.18680-007
45         54 MN      5.02850-006
46         55 CJ      6.57580-007
47         56 NB      3.13320-006
48         57 MO      1.76970-006
49          8    0
50         48 CR      1.48530-003
51         51 NI      1.05570-003
52         52 FE      4.90540-003
53         53 TI      5.14910-007
54         54 MN      1.26440-004
55         55 CJ      3.58570-007
56         56 NB      1.75610-006
57         57 MO      9.91890-007
58         2000.        151.          35.          30.          .30
```

3-LISTING OF INPUT DATA FOR BURNUP SUPER CELL CALCULATION
(ISTO=2)

```
1       ΔASG,A    GE_SCOEFZ.,F24
2       ΔUSE    2,GE_SCOEFZ.
3       ΔASG,T    UGJRFILETWO.,F24
4       ΔUSE   13,UGJRFILETWO.
5       ΔASG,T   UGJRFILETHR.,F24
6       ΔUSE   12,JGJRFILETHR.
7       ΔASG,T   JGJRFILEFOJ.,F24
8       ΔUSE   14,JGJRFILEFOJ.
9       ΔASG,T   JGJRFILEFIV.,F24
10      ΔUSE   22,JGJRFILEFIV.
11      ΔASG,T   JGJRFILESIX.,F24
12      ΔUSE   15,JGJRFILESIX.
13      ΔASG,T   JGJRFILESEV.,F24
14      ΔUSE   16,JGJRFILESEV.
15      ΔASG,T   JGJRFILEEIG.,F24
16      ΔUSE   21,JGJRFILEEIG.
17      ΔASG,A   BJTEK*GELSDATA.
18      ΔUSE   17,BJTEK*GELSDATA.
19      ΔXQT   CSEC,MAIN
20       KWO CLEAN ZERO POWER
21         2  10   2   3   2
22         553.00000   553.00000
23         1   0   0
24       KWO ENRICHMENT   2.5%  SUPER-ZELLE
25         0   0   1   0   0
26             .05104       .04991       .02829
27         1  15  16  45
28         6   4  -1   1  -1   1   1   1  -1   1   2   2   3
29             .47000       .52500       .54600       .68600       .81000      2.9350
30         3   0
31        47 AG 1     2.20990-002
32        48 CD 1     3.34030-004
33        49 IN 1     7.65900-003
34         0   1
35         0   1
36         0   1
37         4   0
38        51 CR 2     1.54335-002
39        57 MN 2     1.39091-003
40        55 FE 2     5.36594-002
41        54 NI 2     8.74816-003
42         0   0
43         4   0
44        51 CR 4     1.68920-002
45        55 FE 4     5.87304-002
46        57 MN 4     1.52236-003
47        54 NI 4     9.57489-003
48         8   0
49        51 CR 5     4.01770-004
50        57 MN 5     9.96750-006
51        55 FE 5     4.81700-004
52        54 NI 5     9.83910-004
53        59 NB 5     6.21050-006
54        58 CO 5     1.30340-006
55        56 TI 5     1.82100-006
56        60 MO 5     3.50730-006
57        12   0
58         5 U5 2.6   1.82791-004
59         9 U8       7.03876-003
60        53 OX 6     1.44430-002
```

```
61          51 CR 6     1.21962-004
62          55 FE 6     1.53299-004
63          54 NI 6     2.78733-004
64          52 ZR 6     4.66819-003
65          57 MN 6     2.81837-006
66          59 NB 6     1.75610-006
67          58 CJ 6     3.68561-007
68          56 TI 6     5.14902-007
69          60 MO 6     9.91882-007
70           0    9
71          2000.          151.          0.          50.          .89
72           5    1   10 .32995
```

4-LISTING OF INPUT DATA FOR SUPER CELL CALCULATION

WITHOUT BURNUP(CONTROL SUPER CELL CAL. ISTO=5)

```
@HDG    SET3.    SUPERCELL
@ASG,T     2.
@ASG,T    12.
@ASG,T    13.
@ASG,T    14.
@ASG,T    15.
@ASG,T    16.
@ASG,T    21.
@ASG,T    22.
@ASG,AX    BUNE*GELSLIB.
@USE    17,BUNE*GELSLIB.
@ASG,AX    BUNE*ABS.
@XQT       BUNE*ABS.GELS
 KWO CLEAN ZERO POWER
   2   4   2   1   4
   553.00000    553.00000
   1
 KWO ENRICHMENT    2.5%     PROZ. ANR.  BRENNSTFF-ZELLE
   0   0   1   1   0
      .05047
   1  15  16  45
   3   2  -1   1  -1   1   1
      .46500       .53700       .81000
   3   0
   5 U5 2.5   5.54650-004
   9 UB       2.13580-002
  53 OX       4.38250-002
   0   1
   4   0
  51 CR       7.60120-005
  52 ZR       4.24540-002
  54 NI       4.75070-006
  55 FE       1.55480-004
   8   0                  /
  51 CR       2.02690-004
  54 NI       4.96380-004
  55 FE       2.43010-004
  56 TI       9.18680-007
  57 MN       5.02850-006
  58 CU       6.57580-007
  59 NB       3.13320-006
  60 MO       1.76970-006
  2000.         151.          0.          50.          .89
 KWO CLEAN ZERO POWER
   2  10   2   3   5
   553.00000    553.00000
   1   0   1
 KWO ENRICHMENT    2.5%  SUPER-ZELLE  PROZ. ANR.
   0   0   1   0   0
   1  10
.05104       .049907      .028287
   1  15  16  45
   6   4  -1   1  -1   1   1   1  -1   1   2   2   3
      .47000       .52500       .64600       .68600       .81000      2.83
   3   0
  47 AG 1     2.20990-002
  48 CD 1     3.34030-004
  49 IN 1     7.65900-003
   0   1
```

```
 0    1
 0    1
 4    0
51 CR 2      1.54335-002
57 MN 2      1.39091-003
55 FE 2      5.36594-002
54 NI 2      8.74816-003
 0    0
 4    0
51 CR 4      1.68920-002
55 FE 4      5.87304-002
57 MN 4      1.52236-003
54 NI 4      9.57489-003
 8    0
51 CR 5      4.01770-004
57 MN 5      9.96760-006
55 FE 5      4.81700-004
54 NI 5      9.83910-004
59 NB 5      6.21050-006
58 CU 5      1.30340-006
56 TI 5      1.82100-006
60 MO 5      3.50780-006
12    0
 5 U5 2.5    1.82791-004
 9 U8        7.03876-003
53 OX 6      1.44430-002
51 CR 6      1.21962-004
55 FE 6      1.53299-004
54 NI 6      2.78733-004
52 ZR 6      4.66819-003
57 MN 6      2.81837-006
59 NB 6      1.75610-006
58 CU 6      3.68561-007
56 TI 6      5.14902-007
60 MO 6      9.91882-007
 0    9
2000.          151.          0.          50.          .89
```

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