ORNL/SUB/98-85B99398V-1

OAK RIDGE
NATIONAL
LABORATORY

LOCKHEED MARTIN

Results of Parametric Design Studies of MOX Lead Test Assembly

Project Manager

A. M. Pavlovitchev

Executed by

S. A. Bychkov
A. A. Lazarenko
V. D. Sidorenko
Y. A. Styrin

MANAGED AND OPERATED BY
LOCKHEED MARTIN ENERGY RESEARCH CORPORATION
FOR THE UNITED STATES
DEPARTMENT OF ENERGY

This report has been reproduced from the best available copy.

Reports are available to the public from the following source.

National Technical Information Service
5285 Port Royal Road
Springfield, VA 22161
Telephone 703-605-6000 (1-800-553-6847)
TDD 703-487-4639
Fax 703-605-6900
E-mail orders@ntis.fedworld.gov
Web site http://www.ntis.gov/ordering.htm

Reports are available to U.S. Department of Energy (DOE) employees, DOE contractors, Energy Technology Data Exchange (ETDE) representatives, and International Nuclear Information System (INIS) representatives from the following source.

Office of Scientific and Technical Information
P.O. Box 62
Oak Ridge, TN 37831
Telephone 423-576-8401
Fax 423-576-5728
E-mail reports@adonis.osti.gov
Web site http://www.osti.gov/products/sources.html

Reports produced after January 1, 1996, are generally available via the DOE Information Bridge.
Web site http://www.doe.gov/bridge

This report was prepared as an account of work sponsored by an agency of the United States Government. Neither the United States Government nor any agency thereof, nor any of their employees, makes any warranty, express or implied, or assumes any legal liability or responsibility for the accuracy, completeness, or usefulness of any information, apparatus, product, or process disclosed, or represents that its use would not infringe privately owned rights. Reference herein to any specific commercial product, process, or service by trade name, trademark, manufacturer, or otherwise, does not necessarily constitute or imply its endorsement, recommendation, or favoring by the United States Government or any agency thereof. The views and opinions of authors expressed herein do not necessarily state or reflect those of the United States Government or any agency thereof.

ORNL/SUB/98-85B99398V-1

RESULTS OF PARAMETRIC DESIGN STUDIES OF MOX LEAD TEST ASSEMBLY

Project Manager

A. M. Pavlovitchev

Executed by

S. A. Bychkov
A. A. Lazarenko
V. D. Sidorenko
Y. A. Styrin

Date Published: December 1998

Report Prepared by
LOCKHEED MARTIN ENERGY RESEARCH CORP.
P.O. Box 2008
Oak Ridge, Tennessee 37831-6363
under
Subcontract Number 85B99398V

Funded by
Office of Fissile Materials Disposition
United States Department of Energy

Prepared for

Computational Physics and Engineering Division
Oak Ridge National Laboratory
Oak Ridge, Tennessee 37831
managed by
LOCKHEED MARTIN ENERGY RESEARCH CORP.
for the
U.S. DEPARTMENT OF ENERGY
under contract DE-96OR22464

Russian Research Center "Kurchatov Institute"
Institute of Nuclear Reactors
VVER Division

Joint U.S. / Russian Project to Update, Verify and Validate
Reactor Design/Safety Computer Codes
Associated with Weapons-Grade Plutonium Disposition in VVER
Reactors

Results of Parametric Design Studies of MOX Lead Test Assembly

(Final Report for FY98)

Project Manager

A.M.Pavlovitchev

Executed by

S.A.Bychkov
A.A. Lazarenko
V.D. Sidorenko
Y.A.Styrin

Moscow 1998

ACRONYMS

BOC	beginning of cycle
BPR	boron poison rod
CR	control rod
EOC	end of cycle
FP	fission products
LTA	lead test assembly
MOX	mixed oxide
SOR	system of regulation
UOX	uranium oxide
VVER	Russian water-water reactor

EXECUTIVE SUMMARY

In this document the results of parametric neutronics studies of MOX LTA design are presented. Two options of MOX LTA design are considered: 100% plutonium and of "island" type. The main part of studies is executed by the Russian code TVS-M.

CONTENTS

INTRODUCTION

This work is a part of Joint U.S. / Russian Project with Weapons-Grade Plutonium Disposition in VVER Reactor and presents the results obtained in the process of parametric studies of MOX LTA design.

The volume and sequence of these studies have been defined in [2] as the stage "Assembly". This report completes the studies partially executed in [3].

At the stage "Assembly" two options of infinite grid are considered:

- grid consisting of single MOX LTAs;
- grid consisting of the following elements: central MOX LTAs surrounded by typical uranium assemblies.

Parametric studies must be resulted in the following features of MOX LTA design:

- Proximity of power generation in MOX LTA and in replaced uranium assembly (Figure 1);
- MOX LTA zoning that ensures acceptable power peaking factor in calculational system.

Two options of MOX LTA are considered within parametric studies:

- 100% plutonium (Figure 3);
- "Island" type (Figure 5, 7).

The Russian cell code TVS-M [3] is used as a calculational instrument. Its main features are evoked in Chapter 2.

In Chapter 3 the calculational model is described.

The results obtained for 100% plutonium option of MOX LTA are presented in Chapter 4, for "island" type option – in Chapter 5.

In the Annex the studies executed in IPPE are presented.

1. Definitions

In the following table the parameters used in current studies are described according to [2].

Parameter	Abbre viation	Units	Remarks
Calculational system	CS		Infinite grid
Reactivity of CS	RO	pcm	RO = (Keff-1)/Keff*1.E5
Effective multiplication factor of CS	Keff		
Multiplication factor of a central assembly in CS	Ko		Relation of neutron generation to neutron absorption.
Average Boron acid (H3BO3) concentration[a] in coolant	Cb	ppm	H3BO3 fraction in coolant (mg of boron acid in 1 Kg of H2O)
Critical boron acid concentration in coolant	Cb crit	ppm	Cb value ensuring Keff=1
2-D power distribution in CS	Kk-CS		Power of fuel pins normalized by average fuel pin power in CS.
Peaking factor of 2-D power distribution in CS	Kkmax-CS		Maximum in Kk-CS values
2-D power distribution in assembly	Kk		Power of fuel pins normalized by average fuel pin power in assembly.
2-D power peaking factor in assembly	(Kk)max		Maximum relative power of fuel pins (maximum of Kk values)
1-D burnup distribution in fuel pin	BUpin		Burnup distribution in concentric zones of equal volume in fuel pin, normalized by average zone burnup.

---

[a] Boron acid concentration divided by the coefficient 5.72 means natural boron (nat B) concentration. In VVER-1000 calculations the term of boron acid concentration is widely used. Below, Cb means boron acid concentration if there is no special indication.

1-D power distribution in fuel pin	qpin		Power distribution in concentric zones of equal volume in fuel pin, normalized by average zone power.
Fission cross section for fast neutrons	Σ_f^{fast}	cm^{-1}	
Fission cross section for thermal neutrons	Σ_f^{th}	cm^{-1}	
Absorption cross section for fast neutrons	Σ_a^{fast}	cm^{-1}	
Absorption cross section for thermal neutrons	Σ_a^{th}	cm^{-1}	
Fast neutron flux	F_1		
Thermal neutron flux	F_2		
Effective fraction of delayed neutrons	βeff		General characteristic of infinite grid
Specific reactor thermal power in CS	Wv	kW/ litre	Reactor thermal power in CS volume unit. For nominal conditions Wv = 108KBt / litre.
Minimum controllable level of reactor power	MCL	MW	In calculations corresponds to Zero Power and uniform temperature 280°C in core.
Average coolant-moderator temperature in CS	T mod	°C	
Average fuel temperature in CS	T fuel	°K	
Average temperature of other CS components	T con	°C	
Xenon-135 concentration distribution	Xe	10*24 /cc	For 1 cc in fuel

Equilibrium Xenon-135 concentration distribution	Xe eq (Wv)	10*24 /cc	Concentration formed during long working with a constant WV.
Sm-149 concentration distribution	Sm	10*24 /cc	For 1 cc in fuel
Equilibrium Sm-149 concentration distribution	Sm eq	10*24 /cc	Concentration formed during long working with constant parameters

2. Short Description of TVS-M

TVS-M is the spectral code for calculations of neutronic constants of cells, super-cells and fuel assemblies of VVER-type reactors. It is a component of code package for VVERs calculations.

A constants library used by TVS-M has the following main features:

- in the fast energy region (E_n> 4.65 KeV) multigroup cross-sections library ABBN is applied. This energy range includes 12 groups of the library. In parallel with the nuclides group constants the subgroup ones are used.
- resonance energy range (4.65 KeV >E_n> 0.625 eV) includes the ABBN groups from 13-th to 24-th (the cross-sections of 24-th group are modified, because the lower boundary of this group is not coincident with the one of the ABBN library). In this energy range the TVS-M code also uses both subgroup and group constants. Besides, the files of resonance parameters from LIPAR-3 library are applied for resonance nuclides. For the most of these nuclides the cross-section calculation is based on the Breit-Wigner multi-level model (and on the Adler-Adler model for fissile nuclides).
- thermal energy range (E_n< 0.625 eV) is subdivided into 24 groups. A set of scattering matrices calculated for various temperatures by the Koppel-Young model is applied for hydrogen bonded in a water molecule. Group cross-sections of nuclides and the scattering matrixes have been obtained with the use of the same algorithms and nuclear data (TEPCON library) as in case of MCU-RFFI/A code.
- 96 fission products are taken into account under burnup calculation. TVS-M code uses library of their yields based on ENDF/B-VI data and group cross-sections from MCU data library.

TVS-M calculation technique consists of the following main stages :

- firstly a detailed calculation of all cell types forming a fuel assembly (such as fuel cell, absorber cell and so on) is performed and corresponding sets of few-group constants are computed (number of the groups is arbitrary)
- then these group effective constants are used in a group nodal diffusion calculation of the whole assembly.

Computing of neutrons spatial distribution in specified energy group (or at specified energy point) is performed by the method of passing through probability (similar to first collision probability method). At the present time an angular distribution of the one-direction neutrons current at a given zone boundary is described by 6 angular harmonics. A neutrons reflection at a cell boundary takes into account a real hexagonal form of the boundary. For a calculation of an effective diffusion coefficient both isotropic and anisotropic probabilities in R and Z directions are computed in the same manner.

In the fast energy region a detailed calculation is carried out with the use of group and subgroup micro cross-sections from the ABBN library. In doing so each energy group is subdivided into arbitrary number of intervals of uniform width. The energy loss of a neutron on non-elastic slowing down is described by continuous function specified by the group matrix of non-elastic transfers. The neutron energy loss on elastic slowing down is also described continuously with taking into account of scattering anisotropy in a system of inertia centre.

In the resonance region the slowing down of neutrons is calculated in the same manner as in the case of fast energy region. Cross-sections of resonance nuclides at each energy point are calculated with the CROSS code using the file of resonance parameters for each nuclide. An interference between potential and resonance scattering, cross-sections temperature dependence, p-wave contribution into scattering cross-section are strictly taken into consideration. An effect of mutual overlapping of different nuclides resonances is also taken into account.

A calculation technique applied in thermal energy region is traditional. The group thermalization equation is solved by the method of passing through probability. The sources are shaped when the upper energy groups are calculated, with the Nelkine asymptotic limit of scattering applied for hydrogen.

Nodal diffusion approach with asymptotic and transient trial functions (both for flux and current) is applied for pin-by-pin calculation of fuel assembly. The asymptotic solution corresponds to the problem with a non-zero source (slowing down or fission) and zero current at the cell boundary. The transient trial function corresponds to the problem on finding neutron distribution in the cell placed at the center of super-cell when a source in it is equal to zero. And in such super-cell a fuel cell is surrounded by the water and a cell of the other type - by homogenized fuel cells. A correction for mesh width is also involved in the balance equation. This correction takes into account the difference between an average flux and a flux at the cell boundary. The similar correction for a current flowed through the cell also appears in the balance equation.

The burnup equations are solved for every fuel pin, which can be subdivided into several concentric rings forming separate burnup zones. Concentration changing of the following heavy nuclides is taken into consideration:

$_{32}$Th^{2} $_{33}$Pa^{2} $_{3}$U^{23} $_{4}$U^{23} $_{5}$U^{23} $_{6}$U^{23} $_{8}$U^{23} $_{237}$Np $_{38}$Pu^{2} $_{239}$Np

$_{39}$Pu^{2} $_{40}$Pu^{2} $_{41}$Pu^{2} $_{42}$Pu^{2} A$_{\mathrm{m}^{241}}$ A$_{\mathrm{m}^{242m}}$ A$_{\mathrm{m}^{243}}$ $_{242}$Cm $_{243}$Cm $_{244}$Cm

Equilibrium concentrations of Xe^{135} and Sm^{149} are also calculated.

3. Calculational Model

Calculational system (CS) for MOX LTA design parametric studies is presented by two principal options:

- infinite grid of single plutonium or uranium assemblies (Fig.);

- infinite grid of central plutonium assemblies surrounded by uranium assemblies of 3.7 %Wt. U-235. The 60° sector of CS for different options of MOX LTA design is shown in Figures 4, 6 and 8. The reference uranium CS is shown in Figure 2.

Composition of weapons grade plutonium is presented in Table 1. The design parameters of plutonium and uranium assemblies are described in Tables 2-5.

The calculational model includes the following two principal regimes described below.

3.1. Fuel Irradiation Simulation

This regime is used for MOX LTA zoning studies under the conditions described in [2]. They comprise irradiation simulation in CS as a rule on the interval [0-40 MWd/kg] with the step 2 MWd/kg.

In the process of irradiation:

- Axial buckling is 1.E-4cm-2. A set of calculations has been executed with a critical buckling ensuring Keff=1;
- Cb (nat B)= 600 ppm;
- Wv = 108 KW/litre;
- Tmod = 302°C;
- Tcon = 302°C;
- Tfuel = 1027°K;
- Xe=Xe eq;
- Sm=Sm eq.

3.2. Zero Power Calculations

This regime is aimed to define reactivity effects due to temperature and Cb variations and to compare Keff with eventual verification calculations to be carried out by other codes.

Calculations are executed in five irradiation points:
0, 10, 20, 30, 40 GWd/t
where states are to be formed by different combinations of the following values:
Cb (nat.B): 0, 600,1200 ppm;
Tmod=Tcon=Tfuel: 20, 280 °C.

4. Calculations of 100 % Plutonium MOX LTA

4.1. Zoning Parametric Studies

Zoning parametric studies consisted in variation of fissile plutonium content in 3-zones MOX LTA (Figure 3).

The results of calculations simulating fuel irradiation in plutonium and uranium assemblies are presented in Tables 7-13. Two options of Uranium reference assembly are considered:

- without BPR i.d. with guide tubes filled by water in 18 positions in assembly (see Figure 1);

- with BPRs of properties presented in Table 6.

It can be seen that 2% fissile plutonium content in periphery (it is the minimum allowable value according to [2]) entails significantly lower values of power peaking factor "Kkmax-CS" than 2.4% content (compare Tables 10 and 11). That is why 2% content in periphery has been adopted. Plutonium content in the central and intermediate zones was variable to obtain Ko value similar to reference uranium CS.

Finally the plutonium content of 4.2/3.0/2.0 has been chosen as acceptable. The Ko evolution in the process of fuel irradiation for the reference uranium and different plutonium assemblies is shown in Figure 10.

Figure 11 shows "Kkmax-CS" evolution in the process of irradiation. The increase of "Kkmax-CS" for 3-zones MOX LTAs is observed from a certain moment. As it is seen from the Table 13 and Figure 9, during irradiation maximum CS power passes from uranium pins out of MOX LTA to the interior of MOX LTA. This effect should be studied in future more attentively taking into account that in real conditions a fresh MOX LTA will be surrounded by both fresh and irradiated uranium assemblies that can lead to mitigating of the mentioned effect.

It is evident that the described procedure of preliminary studies of CS serves only for estimation of eventual performance of MOX LTA in core and that real performance of MOX LTA in core will depend on its real location there. It is quite possible that we should return to the stage "Assembly" after core calculations.

4.2. Zero Power Calculations

The results of calculations are presented in Table 7. It may be seen that the positive temperature reactivity effect appears for the great boron concentrations of 1200 ppm. In MOX LTA this effect is lower owing to more absorbable properties of MOX fuel as compared with uranium one.

5. Calculations of "Island" Type MOX LTA

In these calculations the size of "island" in the center of assembly has been fixed: 54 plutonium fuel pins i.e. 4 pin rows. Two options of "island" have been considered:

- one-zone island or "Island-1"(Figure 5);
- two-zones island or "Island-2"(Figure 7).

The studies are divided into three parts:

- studies of infinite grid of fresh MOX LTA by means of plutonium content variation to ensure acceptable value of power peaking factor Kk. Axial buckling in this case was variable to provide Keff=1.
- calculation of CS, where MOX LTA is surrounded by uranium assemblies, for zoning option chosen in the previous part. In this part plutonium/uranium fuel irradiation has been simulated.
- studies of infinite grid of plutonium assemblies for zoning option chosen in the first part. Axial buckling in this case was variable to provide Keff=1. In this part plutonium/uranium fuel irradiation has been simulated. Inter-pin isotopic and power distributions have been calculated.

The comparison of different spectrum parameters has been also made for a number of combinations of uranium and plutonium fuel enrichments.

Two levels of acceptable values of power peaking factor Kk have been considered:

- Kk=1.20;
- Kk=1.15.

This rather high value of Kk=1.20 was considered in the hope that a proper choice of MOX LTA location in core could lead to rather low power values qi in MOX LTA and finally to an acceptable value of qi*Kk according to safety limits [1, 2].

Uranium zone enrichment inside MOX LTA was equal to 3.7% as a base. In some calculations the option of 4.4% has been also considered.

5.1. "Island-1" option

The studies for uranium zone enrichment of 3.7% have shown (Figure 12) that fissile plutonium content in plutonium zone cannot exceed:

- 2.4% if Kk maximum is 1.15;
- 2.7% if Kk maximum is 1.20.

These values are too low to justify practical using of "Island-1" option in this case.

For uranium zone enrichment of 4.4%, fissile plutonium content in plutonium zone cannot exceed (Figure 13):

- 3.0% if Kk maximum is 1.15;
- 3.4% if Kk maximum is 1.20.

5.2. "Island-2" option

Results of parametric calculations of "Island-2" option have allowed to obtain the pares of plutonium content values in two plutonium zones which could ensure the acceptable value of Kk. The Figures 12 and 13 (correspondingly for uranium zone enrichment of 3.7% and of 4.4%) allow to choose fissile plutonium content ensuring optimum (i.e. minimum) Kk values.

Finally, the chosen zoning is the pair "3.8% in the central part – 2.8% in the island periphery" with uranium environment of 3.7%. In this case, the acceptable power peaking factor, as well as Ko values, close to the reference uranium CS, have been ensured according to Figures 12 and 10.

5.3 "Plutonium island" size variation

Increased size of "Plutonium Island» that comprises 6 plutonium rows (Fig. 14) has been also considered. In Fig. 15 and 16 the central plutonium enrichment has been fixed by 4% while considering two uranium environment enrichments: 3.7% and 4%. The Figures 15 and 16 shows an optimum plutonium periphery enrichment about 3% where Kk minimum is reached.

5.4 Inter-pin isotopic content and power distribution

Inter-pin isotopic content and power distributions are of interest for thermo-hydraulic analysis of MOX fuel behavior. TVS-M allows obtaining of these parameters for 5 concentric zones that have been chosen of equal volumes in current calculations. In Fig.17-28 they are presented for some character pins:

- near central instrumentation tube (as No 77 in Fig.31),
- near water tube (as No 76 in Fig. 31),
- on the border of different "island" enrichments (as No 75 in Fig. 31),
- on the "island" periphery (as No 74 in Fig. 31),
- in uranium fuel pin (as No 72 in Fig. 31).

The following moments while fuel burning have been considered: 12 and 40 MWd/kg that corresponds approximately to fuel discharged after one and three years of reactor exploitation.

Figures 17 and 18 show correspondingly inter-pin relative burnup and power distributions BU_{pin} and q_{pin}. Figures 19-28 show correspondingly inter-pin distribution of U_{235}, PU_{239}, PU_{240}, PU_{241}, PU_{242} for two irradiation levels: 12 and 40 MWd/kg.

5.5 Spectrum characteristics analysis

Usually, more reliable results of treatment of experimental data on fuel pin burning can be obtained if fuel irradiation takes place in the neutron spectrum close to the asymptotic one. It can be seen in Figures 29-31 that in two internal rows of plutonium island "3.8% in the central part – 2.8% in the island periphery" the spectrum is close to the one taking place in 100% Plutonium MOX LTA with the enrichment of 3.8%. So fuel fins located in these positions is reasonable to use for plutonium fuel investigation in the case of "Island" type MOX LTA design.

Relative power distributions are shown in Figures 32 and 33 for the following moments while fuel burning 0,12, 24 and 40 MWd/kg.

Relative burnup distributions are shown in Fig.34 for the following moments while fuel burning 12, 24 and 40 MWd/kg.

Evolution of average assembly neutron absorption and fission cross-sections while fuel burning is presented in Fig.35 for a number of plutonium and uranium enrichment compositions.

Evolution of multiplication factor Ko and power peaking factor Kk while fuel burning is presented in Fig.36 for a number of plutonium and uranium enrichment compositions.

In Figures 37-42 the evolution of U_{235}, PU_{239}, PU_{240}, PU_{241}, PU_{242} and Am_{241} content while fuel burning is presented for a number of plutonium and uranium enrichment compositions.

CONCLUSION

The parametric studies of MOX LTA design have been executed to choose plutonium content in assembly zones for two options of MOX LTA: “3-zones” and “Island”.

For “3-zones” (100% Plutonium) MOX LTA the fissile plutonium content composition of 4.2%/3,0%/2% has been chosen.

MOX LTA of the chosen compositions has been studied by using multi-assembly configuration that allows investigating of influence of MOX LTA environment: uranium assemblies of different irradiation.

Plutonium “Island” with 54 plutonium pins in the center of MOX LTA has been considered in two modifications:

- uniform “island”;
- graded “island” with lower plutonium content in one peripheral row of pins.

It is shown that plutonium content in the uniform “island” cannot exceed 2.7% because of adopted power peaking limitations and therefore this design seems unreasonable for practical use.

For graded “island” the plutonium content composition 3.8%/2.8% with uranium environment of 3.7% U-235 has been chosen.

Evolution of assembly power and burnup distributions, inter-pin power and isotopic distributions while fuel irradiating have been analyzed.

In addition to the base uranium environment of 3,7%, a set of calculations has been executed for 4.4%.

The most of the studies has been executed by the code TVS-M that is at the final stage of licensing and it is to be used in the nearest future as a base instrument for VVER core calculations while using both uranium and MOX fuel. So the obtained results must be considered as preliminary ones and they demand additional analysis and investigations.

REFERENCES

1. Y.A. Styrin. Fuel Assembly and Core Model for Neutronics Calculations of VVER-1000. Draft.
Moscow, Kurchatov Institute 1998.

2. Y.A. Styrin, I.K.Levina. Design of Lead Test MOX Assemblies for Pilot Irradiation in VVER-1000 and Related Parametric Studies. Draft.
Moscow, Kurchatov Institute 1998.

3. S.A. Bichkov, A.P.Lazarenko, V.D.Sidorenko, Y.A. Styrin. Results of Parametric Design Studies of MOX Lead Test Assembly (Progress Report).
Moscow, Kurchatov Institute 1998.

4. V.D.Sidorenko et al. Spectral Code TBC-M for calculation of Characteristics of Cells, Super-cells and Fuel Assemblies of VVER-Type Reactors. 5-th Symposium of the AER.

Table 1. Composition of weapons grade plutonium

Isotope / content (Wt. %)				
Pu-238	Pu-239	Pu-240	Pu-241	Pu-242
0.0	93.0	6.0	1.0	0.0

Table 2. Main Core Parameters

Parameter	Units	Value
Thermal Power	MW thermal	3000
Electrical Power	MW	1000
Number of Coolant Loops		4
Number of Fuel Assemblies		163
Core Equivalent Diameter	m	3.164
Core Fuel Height	m	3.53
Core Volume	m^3	27.8
Core Power Density	W/cm^3	108
Control / Shut off Rod Banks		10
Position of Regulating Rod Bank	%	90
Core Coolant Flow Rate	m^3/hr	84000
Pressure at Core Inlet	MPa	15.7
Core Inlet Temperature	°C	287

Table 3. Fuel Assembly Design Parameters

Parameter	Units	Value
Shape of Fuel Assembly		Hexagonal
Distance Across Assembly (between flats)	cm	23.4
Distance Between Fuel Assembly Centres	cm	23.6
Fuel Pin Lattice Pitch	cm	1.275
Number of Fuel Pins in Fuel Assembly		312
Number of Guide Tubes for Control Rods / Burnable Absorber Pins		18
Inner Diameter of Guide Thimbles	cm	1.1
Thickness of Guide Thimbles	cm	0.1
Material of Guide Thimbles		Zirconium Alloy*
Central Instrumentation Tube Inner Diameter	cm	1.1
Thickness of Central Instrumentation Tube	cm	0.1
Material of Central Guide Tube		Zirconium Alloy *
Number of Spacer Grids in Fuel Assembly		13
Material of Spacer Grids		Zirconium Alloy*
Spacer Grid Weight (each)	Kg	0.55

Compositions Weight percent:
*

Zr	Nb	Hf
98.97	1.0	0.03

Table 4. Uranium Fuel Pin Design Parameters

Parameter	Units	Value
		Advanced Core Design
Inner Clad Diameter	cm	0.772
Clad Thickness	cm	0.069
Clad Material		Zirconium Alloy*
Clad Density	g / cc	6.5153
Fuel Pellet Diameter	cm	0.755
Central Hole Diameter	cm	0.15
Fuel Pellet Material		L.E. UO2
Height of Fuel Column	cm	353 (cold) 355 (hot)
Mass of UO2 in Fuel Pin	kg	1.575

Compositions Weight percent:

*

Zr	Nb	Hf
98.97	1.0	0.03

Table 5. MOX fuel Pin Design Parameters

Parameter	Units	Value
Inner Clad Diameter	cm	0.772
Clad Thickness	cm	0.069
Clad Material		Zirconium Alloy*
Clad Density	g / cc	6.5153
Fuel Pellet Diameter	cm	0.755
Central Hole Diameter	cm	0.15
U-235 content in MOX fuel	%	0.2
Fuel Pellet Material		PuO2-UO2
Height of Fuel Column	cm	353 (cold) 355 (hot)
Fuel Density	g / cc	10.5

Compositions Weight percent:

*

Zr	Nb	Hf
98.97	1.0	0.03

Table 6. Discrete Burnable Poison Pin Design Parameters

Parameter	Units	Value
Clad Inner Diameter	cm	0.772
Clad Thickness	cm	0.069
Clad Material		Zirconium Alloy*
Clad Density	g / cc	6.5153
Absorber Diameter	cm	0.758
Absorber Density	g / cc	2.945
Absorber Composition		Boron g / cc
		0.065
B10	Wt%	0.4046
B11		1.8028
Al		88.5951
Fe		0.1850
Ni		1.8496
Cr		5.3133
Zr		1.8496

Compositions Weight percent:

*

Zr	Nb	Hf
98.97	1.0	0.03

Table 7. Keff in Zero Power States

Irradiation Point →	0				10, GWd/t				20, GWd/t				30, GWd/t				40, GWd/t			
	Tmod=Tfuel =Tcon =20°C		Tmod=Tfuel =Tcon =280°C		Tmod=Tfuel =Tcon =20°C		Tmod=Tfuel =Tcon =280°C		Tmod=Tfuel =Tcon =20°C		Tmod=Tfuel =Tcon =280°C		Tmod=Tfuel =Tcon =20°C		Tmod=Tfuel =Tcon =280°C		Tmod=Tfuel =Tcon =20°C		Tmod=Tfuel =Tcon =280°C	
Cb (nat.B) → Pu/U Content, % ↓	0	1200	0	1200	0	1200	0	1200	0	1200	0	1200	0	1200	0	0	1200	0	1200	1200
U: 3.7/3.3 no BPR	1.4390	1.2266	1.3965	1.2370	1.2731	1.0952	1.2295	1.1028	1.1815	1.0134	1.1397	1.0221	1.0982	0.9374	1.0620	0.9501	1.0170	0.8637	0.9869	0.8802
U: 3.7/3.3 with BPR	1.4010	1.1991	1.3513	1.2015	1.2484	1,0786	1.2019	1.0817	1.1683	1.0061	1.1244	1.0113	1.0905	0.9345	1.0517	0.9436	1.0155	0.8660	0.9839	0.8802
PU: 4.4/3.0/2.4	1.4242	1.2341	1.3725	1.2342	1.2576	1.0967	1.2102	1.0979	1.1689	1.0152	1.1245	1.0188	1.0896	0.9414	1.0510	0.9492	1.0174	0.8742	0.9858	0.8872
PU: 4.4/3.0/2.0	1.4233	1.2322	1.3720	1.2331	1.2563	1.0945	1.2093	1.0963	1.1674	1.0130	1.1234	1.0171	1.0880	0.9392	1.0498	0.9476	1.0157	0.8720	0.9846	0.8855
PU: 4.4/3.2/2.0	1.4237	1.2331	1.3724	1.2337	1.2572	1.0958	1.2100	1.0974	1.1685	1.0144	1.1243	1.0183	1.0891	0.9407	1.0508	0.9488	1.0169	0.8735	0.9856	0.8868
PU: 4.2/3.0/2.0	1.42291	1.2315	1.3717	1.2325	1.2536	10920	1.2066	1.0940	1.1636	1.0096	1.1197	1.0138	1.0837	0.9354	1.0457	0.9440	1.0114	0.8683	0.9805	0.8820
PU-Island: 3.8/2.8/U-3.7	1.4328	1.2261	1.3861	1.2325	1.2652	1.0922	1.2189	1.0966	1.1738	1.0101	1.1296	1.0159	1.0914	0.9347	1.0530	0.9446	1.0157	0.8653	0.9847	0.8805

Table 8. Parameters Evolution in the Process of Fuel Irradiation. Reference Uranium Assemblage. No BPR

Irradiation Point →	Burnup, GWd/t																				
Parameters ↓	0	2	4	6	8	10	12	14	16	18	20	22	24	26	28	30	32	34	36	38	40
Keff	1.2358	1.2168	1.1971	1.1768	1.1569	1.1378	1.1194	1.1018	1.0848	1.0684	1.0525	1.0431	1.0219	1.0071	0.9927	0.9786	0.9648	0.9513	0.9381	0.9252	0.9126
Ko	1.2402	1.2212	1.2014	1.1809	1.1608	1.1415	1.1230	1.1052	1.0881	1.0715	1.0555	1.0398	1.0246	1.0097	0.9951	0.9809	0.9669	0.9534	0.9401	0.9271	0.9145
Kkmax-CS	1.0740 (46)	1.0726 (46)	1.0708 (46)	1.0688 (46)	1.0664 (46)	1.0642 (46)	1.0619 (46)	1.0594 (46)	1.0565 (46)	1.0539 (46)	1.0514 (46)	1.0486 (46)	1.0460 (46)	1.0431 (46)	1.0407 (46)	1.0378 (46)	1.0353 (46)	1.0329 (46)	1.0305 (46)	1.0284 (46)	1.0262 (46)
βeff	0.007197	0.006915	0.006668	0.006463	0.006287	0.006133	0.005996	0.005873	0.005762	0.005660	0.005567	0.005480	0.005399	0.005323	0.005252	0.005184	0.005121	0.005061	0.005003	0.004949	0.004897

Table 9. Parameters Evolution in the Process of Fuel Irradiation. Reference Uranium Assemblage with BPR

Irradiation Point →	Burnup, GWd/t																				
Parameters ↓	0	2	4	6	8	10	12	14	16	18	20	22	24	26	28	30	32	34	36	38	40
Keff	1.2047	1.1883	1.1712	1.1536	1.1364	1.1199	1.1104	1.0890	1.0742	1.0597	1.0454	1.0312	1.0171	1.0031	0.9893	0.9756	0.9622	0.9490	0.9360	0.9234	0.9111
Ko	1.1113	1.1076	1.1029	1.0970	1.0907	1.0844	1.0780	1.0712	1.0637	1.0555	1.0462	1.0359	1.0248	1.0130	1.0007	0.9881	0.9754	0.9628	0.9502	0.9378	0.9257
Kkmax-CS	1.1289 (46)	1.1213 (46)	1.1136 (46)	1.1059 (46)	1.0983 (46)	1.0907 (46)	1.0834 (46)	1.0763 (46)	1.0697 (46)	1.0635 (46)	1.0579 (46)	1.0528 (46)	1.0483 (46)	1.0442 (46)	1.0405 (46)	1.0371 (46)	1.0339 (46)	1.0310 (46)	1.0283 (46)	1.0258 (46)	1.0234 (46)
βeff	0.007199	0.006911	0.006660	0.006451	0.006273	0.006118	0.005982	0.005859	0.005748	0.005647	0.005554	0.005468	0.005388	0.005314	0.005243	0.005177	0.005115	0.005056	0.005000	0.004946	0.004895

Table 10. Parameters Evolution in the Process of Fuel Irradiation. MOX LTA 4.4/3.0/2.4

Irradiation Point →	Burnup, GWd/t																				
Parameters ↓	0	2	4	6	8	10	12	14	16	18	20	22	24	26	28	30	32	34	36	38	40
Keff	1.2313	1.2100	1.1895	1.1693	1.1497	1.1311	1.1133	1.0962	1.0798	1.0640	1.0487	1.0338	1.0193	1.0051	0.9913	0.9779	0.9647	0.9519	0.9394	0.9272	0.9038
Ko	1.2439	1.2170	1.1911	1.1698	1.1502	1.1318	1.1143	1.0977	1.0818	1.0665	1.0518	1.0375	1.0237	1.0104	0.9975	0.9849	0.9727	0.9609	0.9495	0.9384	0.9172
Kkmax-CS	1.1898	1.1394	1.0979	1.0924	1.0878	1.0831	1.0781	1.0737	1.0751	1.0798	1.0841	1.0881	1.0917	1.0950	1.0980	1.1006	1.1029	1.1049	1.1065	1.1079	1.1112

Table 11. Parameters Evolution in the Process of Fuel Irradiation. MOX LTA 4.4/3.0/2.0

Irradiation Point →	Burnup, GWd/t																				
Parameters ↓	0	2	4	6	8	10	12	14	16	18	20	22	24	26	28	30	32	34	36	38	40
Keff	1.2306	1.2091	1.1886	1.1682	1.1487	1.1300	1.1121	1.0951	1.0786	1.0628	1.0474	1.0325	1.0180	1.0039	0.9901	0.9766	0.9634	0.9506	0.9381	0.9259	0.9140
Ko	1.2384	1.2090	1.1853	1.1640	1.1444	1.1260	1.1087	1.0923	1.0765	1.0614	1.0469	1.0329	1.0193	1.0061	0.9934	0.9811	0.9691	0.9575	0.9461	0.9353	0.9247
Kkmax-CS	1.1027 (46)	1.1016 (46)	1.0984 (46)	1.0941 (46)	1.0896 (46)	1.0849 (46)	1.0802 (46)	1.0760 (46)	1.0810 (206)	1.0856 (206)	1.0896 (206)	1.0936 (206)	1.0971 (206)	1.1002 (206)	1.1031 (136)	1.1056 (136)	1.1076 (136)	1.1091 (136)	1.1103 (206)	1.1114 (206)	1.1122 (206)
βeff	0.005978	0.005804	0.005650	0.005523	0.005415	0.005322	0.005241	0.005170	0.005107	0.005050	0.004998	0.004951	0.004908	0.004868	0.004831	0.004796	0.004763	0.004733	0.004704	0.004676	0.004650

Table 12. Parameters Evolution in the Process of Fuel Irradiation. MOX LTA 4.4/3.2/2.0

Irradiation Point →	Burnup, GWd/t																				
Parameters ↓	0	2	4	6	8	10	12	14	16	18	20	22	24	26	28	30	32	34	36	38	40
Keff	1.2311	1.2098	1.1893	1.1690	1.1495	1.1308	1.1130	1.0960	1.0796	1.0638	1.0484	1.0335	1.0190	1.0049	0.9911	0.9776	0.9645	0.9517	0.9391	0.9269	0.9151
Ko	1.2407	1.2116	1.1881	1.1670	1.1475	1.1293	1.1121	1.0957	1.0800	1.0649	1.0504	1.0364	1.0228	1.0097	0.9969	0.9846	0.9725	0.9609	0.9495	0.9385	0.9278
Kkmax-CS	1.1020 (46)	1.1007 (46)	1.0974 (46)	1.0931 (46)	1.0885 (46)	1.0837 (46)	1.0787 (46)	1.0740 (46)	1.0718 (206)	1.0768 (206)	1.0815 (206)	1.0857 (206)	1.0896 (206)	1.0932 (206)	1.0964 (206)	1.0994 (206)	1.1019 (206)	1.1042 (206)	1.1061 (206)	1.1077 (206)	1.1090 (206)
βeff	0.005972	0.005798	0.005645	0.005518	0.005410	0.005318	0.005237	0.005165	0.005102	0.005045	0.004993	0.004946	0.004903	0.004864	0.004827	0.004792	0.004760	0.004730	0.004701	0.004674	0.004648

Table 13. Parameters Evolution in the Process of Fuel Irradiation. MOX LTA 4.2/3.0/2.0

Irradiation Point →	Burnup, GWd/t																				
Parameters ↓	0	2	4	6	8	10	12	14	16	18	20	22	24	26	28	30	32	34	36	38	40
Keff	1.2300	1.2081	1.1871	1.1664	1.1465	1.1275	1.1094	1.0921	1.0755	1.0595	1.0440	1.0290	1.0144	1.0002	0.9863	0.9728	0.9596	0.9468	0.9343	0.9221	0.9103
Ko	1.2361	1.2062	1.1820	1.1603	1.1403	1.1217	1.1041	1.0874	1.0714	1.0560	1.0413	1.0271	1.0134	1.0001	0.9872	0.9748	0.9627	0.9511	0.9398	0.9288	0.9182
Kkmax-CS	1.1043 (46)	1.1030 (46)	1.0997 (46)	1.0955 (46)	1.0910 (46)	1.0863 (46)	1.0816 (46)	1.0769 (46)	1.0723 (46)	1.0693 (206)	1.0734 (206)	1.0770 (206)	1.0803 (206)	1.0833 (206)	1.0859 (206)	1.0882 (206)	1.0902 (206)	1.0919 (206)	1.0933 (206)	1.0944 (206)	1.0963 (253)
βeff	0.005985	0.005810	0.005656	0.005529	0.005421	0.005329	0.005248	0.005177	0.005113	0.005056	0.005005	0.004957	0.004914	0.004874	0.004837	0.004803	0.004763	0.004739	0.004710	0.004682	0.004656

Table 14. Parameters Evolution in the Process of Fuel Irradiation. MOX LTA 3.8/2.8/U-3.7

Irradiation Point →	Burnup, GWd/t																				
Parameters ↓	0	2	4	6	8	10	12	14	16	18	20	22	24	26	28	30	32	34	36	38	40
Keff	1.2357	1.2156	1.1953	1.1747	1.1547	1.1354	1.1170	1.0994	1.0824	1.0660	1.0502	1.0347	1.0197	1.0051	0.9908	0.9768	0.9631	0.9498	0.9368	0.9241	0.9117
Ko	1.2409	1.2190	1.1984	1.1780	1.1582	1.1394	1.1214	1.1040	1.0873	1.0712	1.0555	1.0403	1.0255	1.0111	0.9969	0.9832	0.9697	0.9565	0.9436	0.9311	0.9189
Kkmax-CS	1.2064 (210)	1.1890 (210)	1.1785 (210)	1.1711 (210)	1.1649 (210)	1.1592 (210)	1.1532 (210)	1.1472 (210)	1.1409 (210)	1.1345 (210)	1.1279 (210)	1.1211 (210)	1.1144 (210)	1.1077 (230)	1.1011 (230)	1.0984 (231)	1.0964 (275)	1.0963 (253)	1.0958 (253)	1.0949 (253)	1.0938 (253)
βeff	0.006934	0.006681	0.006459	0.006274	0.006115	0.005976	0.005853	0.005743	0.005643	0.005552	0.005468	0.005390	0.005318	0.005250	0.005186	0.005126	0.005069	0.005015	0.004964	0.004915	0.004868

Figure 1. Simplified Design for Uranium Reference Assembly

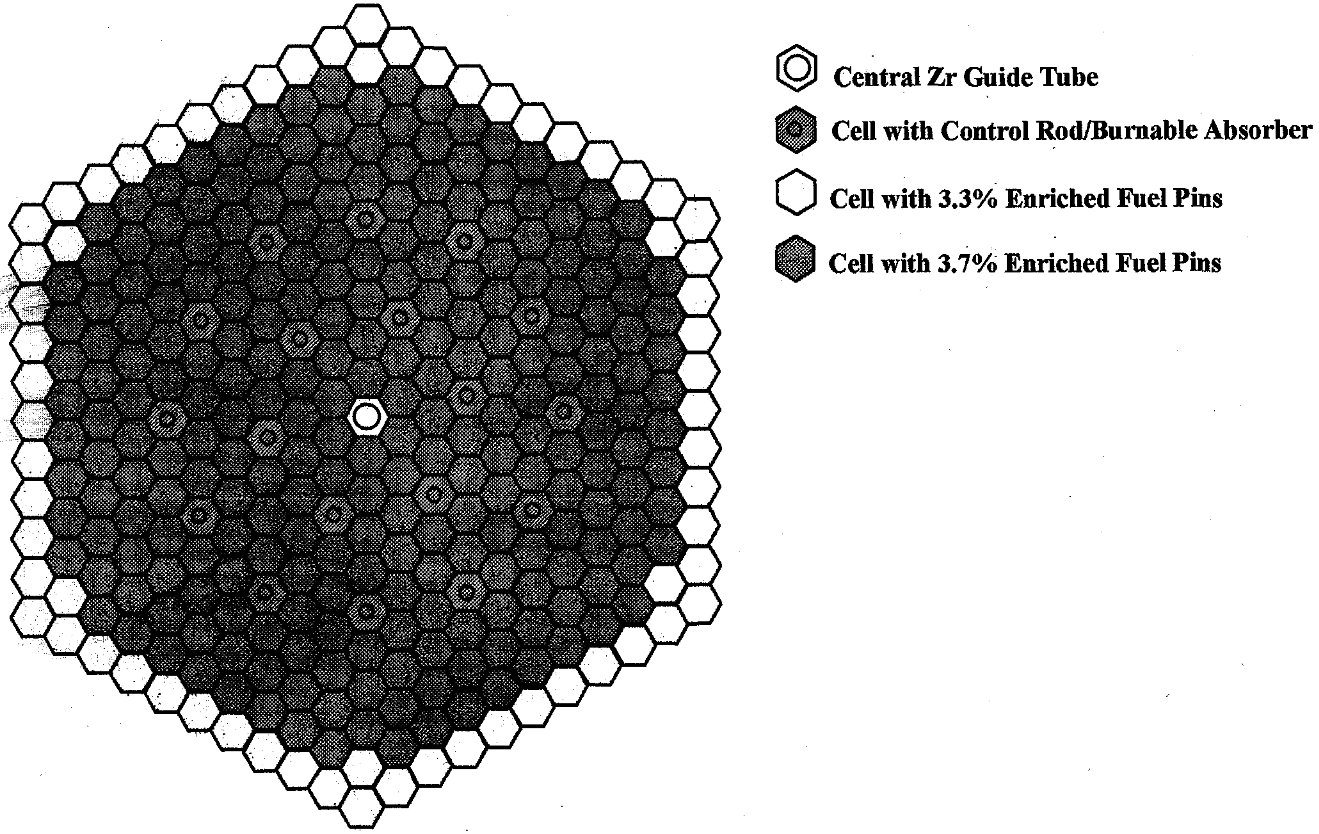
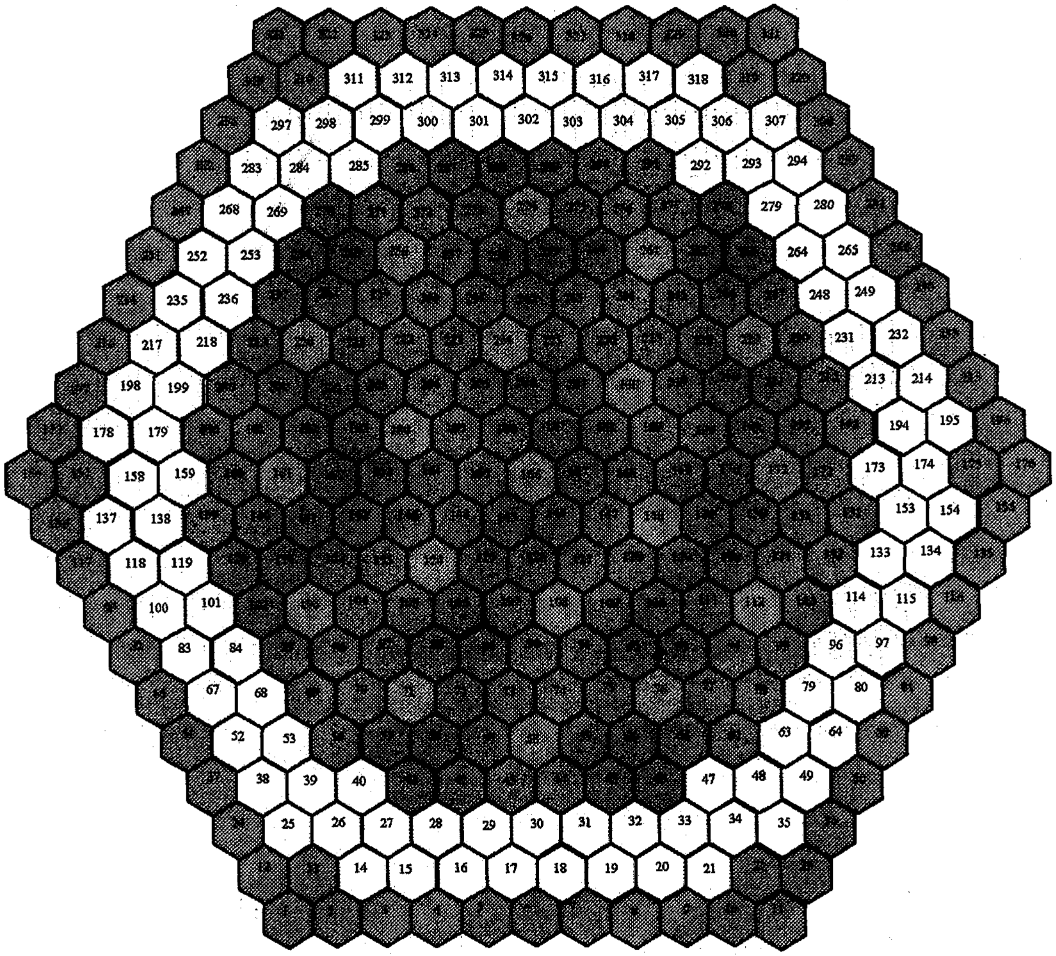


Figure 2. Calculational Model for Reference Uranium Assembly Surrounded by Uranium Assemblies. 60° Sector

```
26,
71,25,
71,71,25,
71,71,71,25,
71,71,71,71,25,
71,71,71,71,71,25,
29,71,71,71,71,71,25,
71,71,71,71,71,71,71,25,
71,71,71,29,71,71,71,71,25,
71,29,71,71,71,71,71,71,71,25,
71,71,71,71,71,71,71,71,71,71,25,
27,71,71,71,71,29,71,71,71,71,71,26,
71,71,71,29,71,71,71,71,71,71,71,25,64,
71,71,71,71,71,71,71,71,71,71,71,25,64,64,
71,71,29,71,71,71,29,71,71,71,71,25,64,50,50,
71,71,71,71,71,71,71,71,71,71,71,25,64,50,50,50,
29,71,71,71,71,29,71,71,71,71,71,25,64,50,50,50,50,
71,71,71,29,71,71,71,71,71,71,71,25,64,50,50,50,50,29,
71,71,71,71,71,71,71,71,71,71,71,25,64,50,50,50,50,50,50,
71,71,71,71,71,71,71,71,71,71,71,25,64,50,50 50,29,50,50,50,
71,71,71,71,71,71,71,71,71,71,71,25,64,50,50,50,50,50,50,50,50,
71,71,71,71,71,71,71,71,71,71,71,25,64,50,50,50,50,50,50,29,50,50,
26,25,25,25,25,25,25,25,25,25,25,26,64,64,50,50,50,29,50,50,50,50,27,
```

25 – side water cell
26 – corner water cell
27 – central tube cell
29 – guide tube cell / burnable absorbers
50 – uranium 3.7% U-235 fuel rods
64 – uranium 3.3% U-235 fuel rods
71 – uranium 3.7% U-235 fuel rods

Figure 3. Simplified Design for 3 Zones MOX LTA



⬢ **Low Plutonium-Content MOX Rods**

⬡ **Intermediate Plutonium-Content MOX Rods**

⬢ **High Plutonium-Content MOX Rods**

⬢ **Central tube**

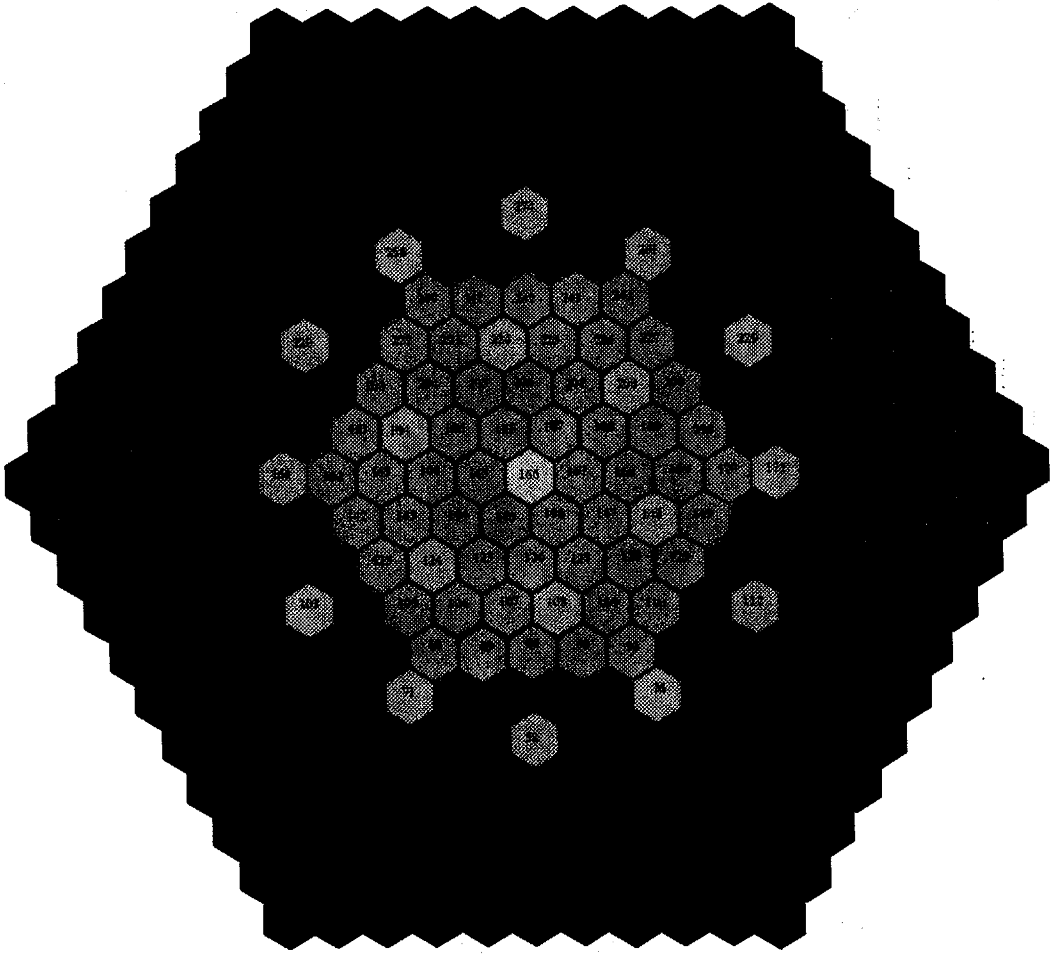
⬢ **Control Rods / Burnable Absorbers**

Figure 4. Calculational Model for 3-Zones (100 % Plutonium) MOX LTA Surrounded by Uranium Assemblies. 60° Sector

```
                         26,
                        71,25,
                       71,71,25,
                      71,71,71,25,
                     71,71,71,71,25,
                    71,71,71,71,71,25,
                   29,71,71,71,71,71,25,
                  71,71,71,71,71,71,71,25,
                 71,71,71,29,71,71,71,71,25,
                71,29,71,71,71,71,71,71,71,25,
               71,71,71,71,71,71,71,71,71,71,25,
              27,71,71,71,71,29,71,71,71,71,71,26,
             71,71,71,29,71,71,71,71,71,71,71,25,64,
            71,71,71,71,71,71,71,71,71,71,71,25,64,64,
           71,71,29,71,71,71,29,71,71,71,71,25,64,        ,
          71,71,71,71,71,71,71,71,71,71,71,25,64,
         29,71,71,71,71,29,71,71,71,71,71,25,64,      ,50,50,
        71,71,71,29,71,71,71,71,71,71,71,25,64,      ,50,50,29,
       71,71,71,71,71,71,71,71,71,71,71,25,64,      ,50,50,50,50,
      71,71,71,71,71,71,71,71,71,71,71,25,64,      50,29,50,50,50,
     71,71,71,71,71,71,71,71,71,71,71,25,64,      ,50,50,50,50,50,50,
    71,71,71,71,71,71,71,71,71,71,71,25,64,      ,50,50,50,50,29,50,50,
   26,25,25,25,25,25,25,25,25,25,25,26,64,64,      ,50,29,50,50,50,50,27,
```

25 – side water cell
26 – corner water cell
27 – central tube cell
29 – guide tube cell / burnable absorbers
50 – high plutonium-content fuel rods
– intermediate plutonium-content fuel rods
64 – low plutonium-content fuel rods
71 – uranium 3.7% U-235 fuel rods

Figure 5. Simplified Design for "Island-1" Type MOX LTA



⬢ **Enriched Uranium Rods**

⬢ **Central tube**

⬢ **High Plutonium-Content MOX Rods**

⬢ **Control Rods / Burnable Absorbers**

Figure 6. Calculational Model for "Island-1" MOX LTA Surrounded by Uranium Assemblies. 60° Sector

```
                                26,
                               71,25,
                              71,71,25,
                             71,71,71,25,
                            71,71,71,71,25,
                           71,71,71,71,71,25,
                          29,71,71,71,71,71,25,
                         71,71,71,71,71,71,71,25,
                        71,71,71,29,71,71,71,71,25,
                       71,29,71,71,71,71,71,71,71,25,
                      71,71,71,71,71,71,71,71,71,71,25,
                     27,71,71,71,71,29,71,71,71,71,71,26,
                    71,71,71,29,71,71,71,71,71,71,71,25,64,
                   71,71,71,71,71,71,71,71,71,71,71,25,64,64,
                  71,71,29,71,71,71,29,71,71,71,71,25,64,
                 71,71,71,71,71,71,71,71,71,71,71,25,64,
                29,71,71,71,71,29,71,71,71,71,71,25,64,
               71,71,71,29,71,71,71,71,71,71,71,25,64,              ,28,
              71,71,71,71,71,71,71,71,71,71,71,25,64,                ,50,
             71,71,71,71,71,71,71,71,71,71,71,25,64,           ,28,  ,50,50,
            71,71,71,71,71,71,71,71,71,71,71,25,64,                 ,50,50,50,
           71,71,71,71,71,71,71,71,71,71,71,25,64,                ,50,28,50,50,
          26,25,25,25,25,25,25,25,25,25,25,26,64,64,           ,28,50,50,50,50,27,
```

25 – side water cell
26 – corner water cell
27 – central tube cell
28, 29 – guide tube cell / burnable absorbers
50 –plutonium fuel rods
– uranium 3.7% U-235 fuel rods
64 – uranium 3.3% U-235 fuel rods
71 – uranium 3.7% U-235 fuel rods

Figure 7. Simplified Design for "Island-2" Type MOX LTA

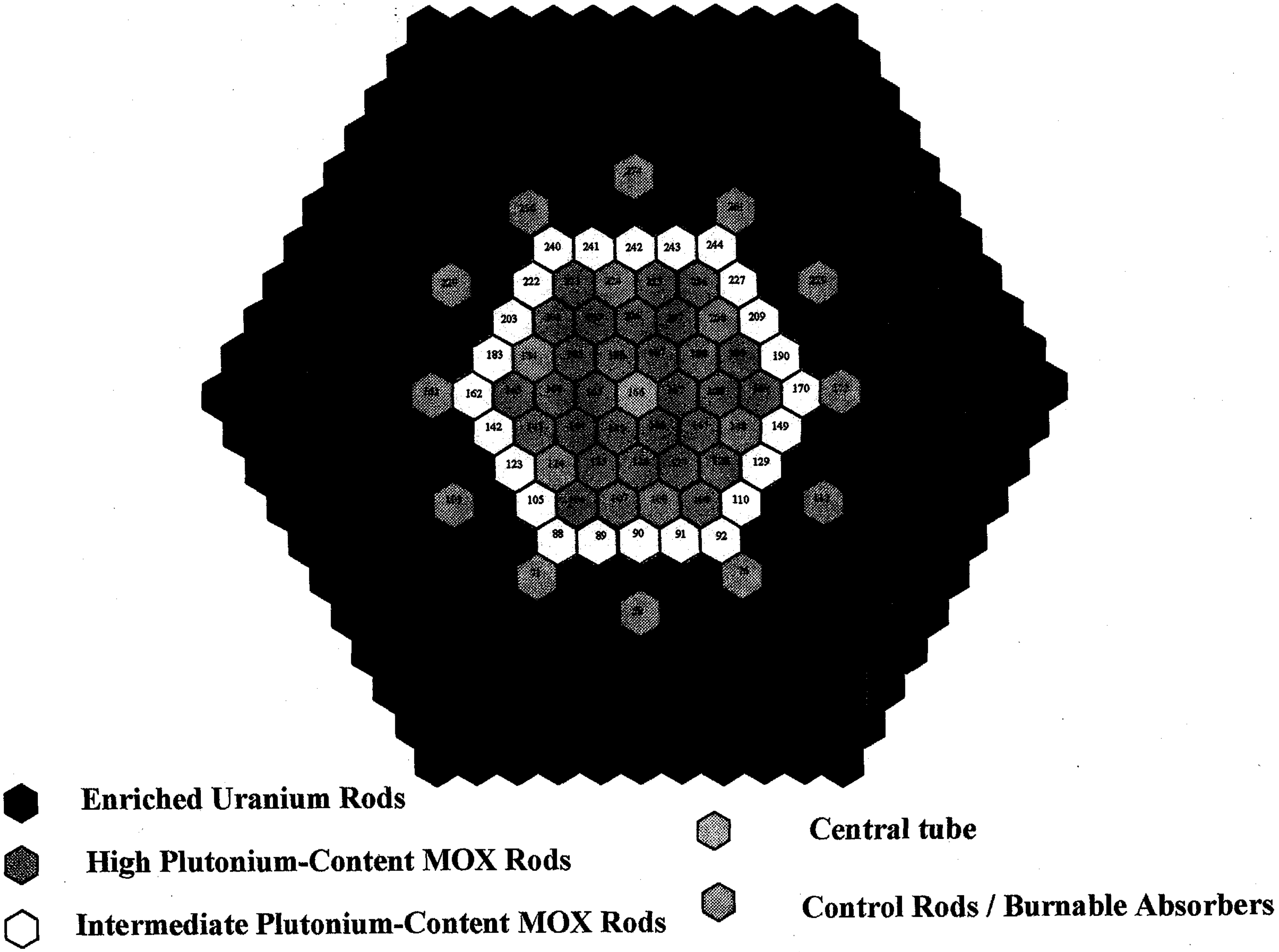


Figure 8. Calculational Model for "Island-2" MOX LTA Surrounded by Uranium Assemblies. 60° Sector

```
                                26,
                               71,25,
                              71,71,25,
                             71,71,71,25,
                            71,71,71,71,25,
                           71,71,71,71,71,25,
                          29,71,71,71,71,71,25,
                         71,71,71,71,71,71,71,25,
                        71,71,71,29,71,71,71,71,25,
                       71,29,71,71,71,71,71,71,71,25,
                      71,71,71,71,71,71,71,71,71,71,25,
                     27,71,71,71,71,29,71,71,71,71,71,26,
                    71,71,71,29,71,71,71,71,71,71,71,25,
                   71,71,71,71,71,71,71,71,71,71,71,25,
                  71,71,29,71,71,71,29,71,71,71,71,25,
                 71,71,71,71,71,71,71,71,71,71,71,25,
                29,71,71,71,71,29,71,71,71,71,71,25,
               71,71,71,29,71,71,71,71,71,71,71,25,                  ,28,
              71,71,71,71,71,71,71,71,71,71,71,25,                     ,64,
             71,71,71,71,71,71,71,71,71,71,71,25,               ,28,   ,64,64,
            71,71,71,71,71,71,71,71,71,71,71,25                      ,64,50,50,
           71,71,71,71,71,71,71,71,71,71,71,25,                     ,64,28,50,50,
          26,25,25,25,25,25,25,25,25,25,25,26,                  ,28,64,64,50,50,27,
```

25 – side water cell
26 – corner water cell
27 – central tube cell
28, 29 – guide tube cell / burnable absorbers
50 –high plutonium fuel rods
– uranium 3.7% U-235 fuel rods
64 – low plutonium fuel rods
71 – uranium 3.7% U-235 fuel rods

Figure 9. Pins Numeration in CS Model

**1 ,
2 , 3 ,
4, 5, 6,
7, 8, 9, 10,
11, 12, 13, 14, 15,
16, 17, 18, 19, 20, 21,
22, 23, 24, 25, 26, 27, 28,
29, 30, 31, 32, 33, 34, 35, 36,
37, 38, 39, 40, 41, 42, 43, 44, 45,
46, 47, 48, 49, 50, 51, 52, 53, 54, 55,
56, 57, 58, 59, 60, 61, 62, 63, 64, 65, 66,
67, 68, 69, 70, 71, 72, 73, 74, 75, 76, 77, 78,
79, 80, 81, 82, 83, 84, 85, 86, 87, 88, 89, 90, 91,
92, 93, 94, 95, 96, 97, 98, 99, 100,101,102,103,104,105,
106,107,108,109,110,111,112,113,114,115,116,117,118,119,120,
121,122,123,124,125,126,127,128,129,130,131,132,133,134,135,136,
137, 138,139,140,141,142,143,144,145,146,147,148,149,150,151,152,153,
154,155,156,157,158,159,160,161,162,163,164,165,166,167,168,169,170,171,
172,173,174,175,176,177,178,179,180,181,182,183,184,185,186,187,188,189,190,
191,192,193,194,195,196,197,198,199,200,201,202,203,204,205,206,207,208,209,210,
211,212,213,214,215,216,217,218,219,220,221,222,223,224,225,226,227,228,229,230,231,
232,233,234,235,236,237,238,239,240,241,242,243,244,245,246,247,248,249,250,251,252,253,
254,255,256,257,258,259,260,261,262,263,264,265,266,267,268,269,270,271,272,273,274,275,276,**

**257 – side water cell
254 – corner water cell
276 – central tube cell
137 – guide tube cell / burnable absorbers
223 –plutonium fuel rods
71 – uranium 3.7% U-235 fuel rods**

Figure 10. Evolution of Ko in Plutonium-Uranium Super-Cells

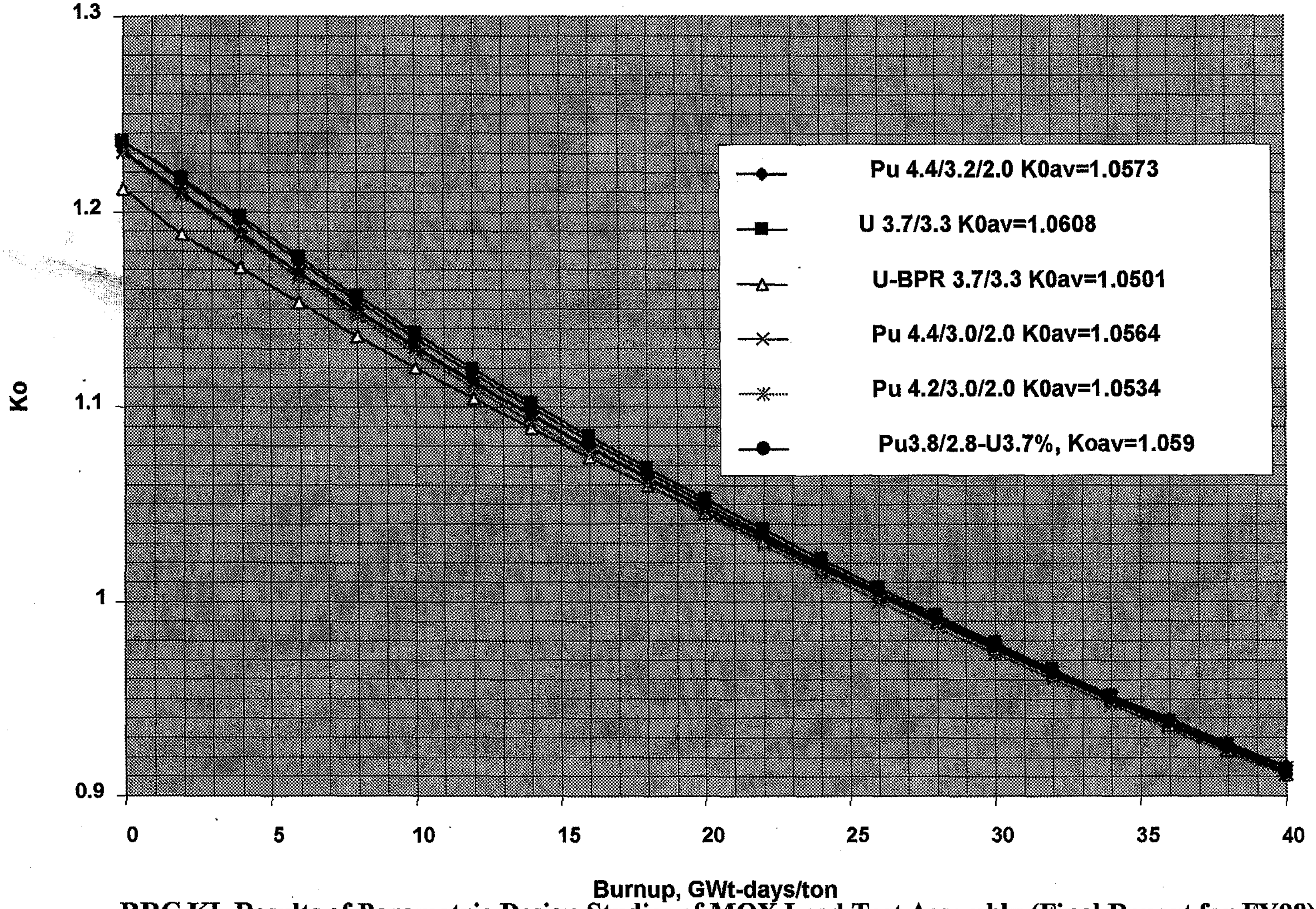


Figure 11. Evolution of Kk in Plutonium-Uranium Super-Cells

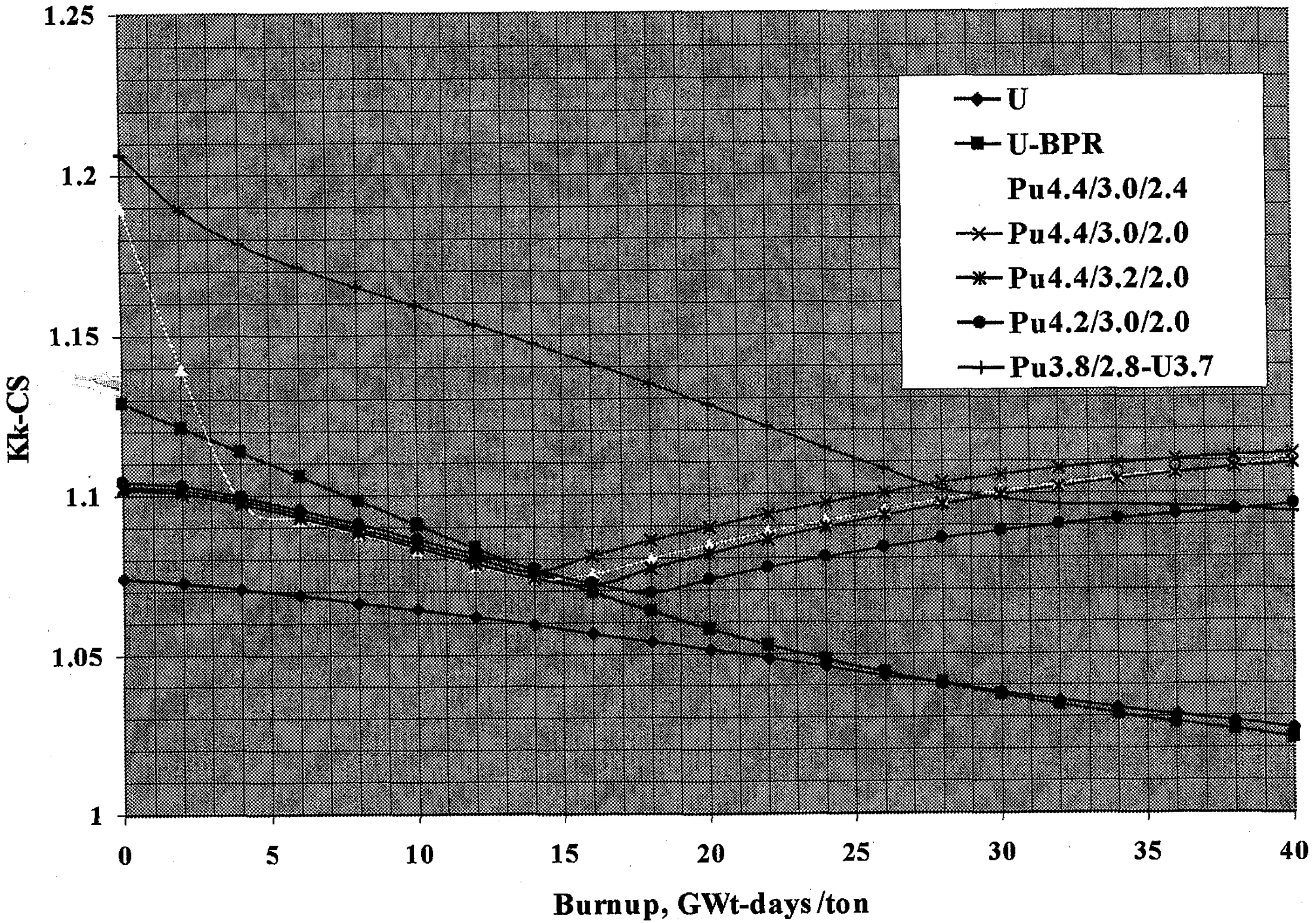
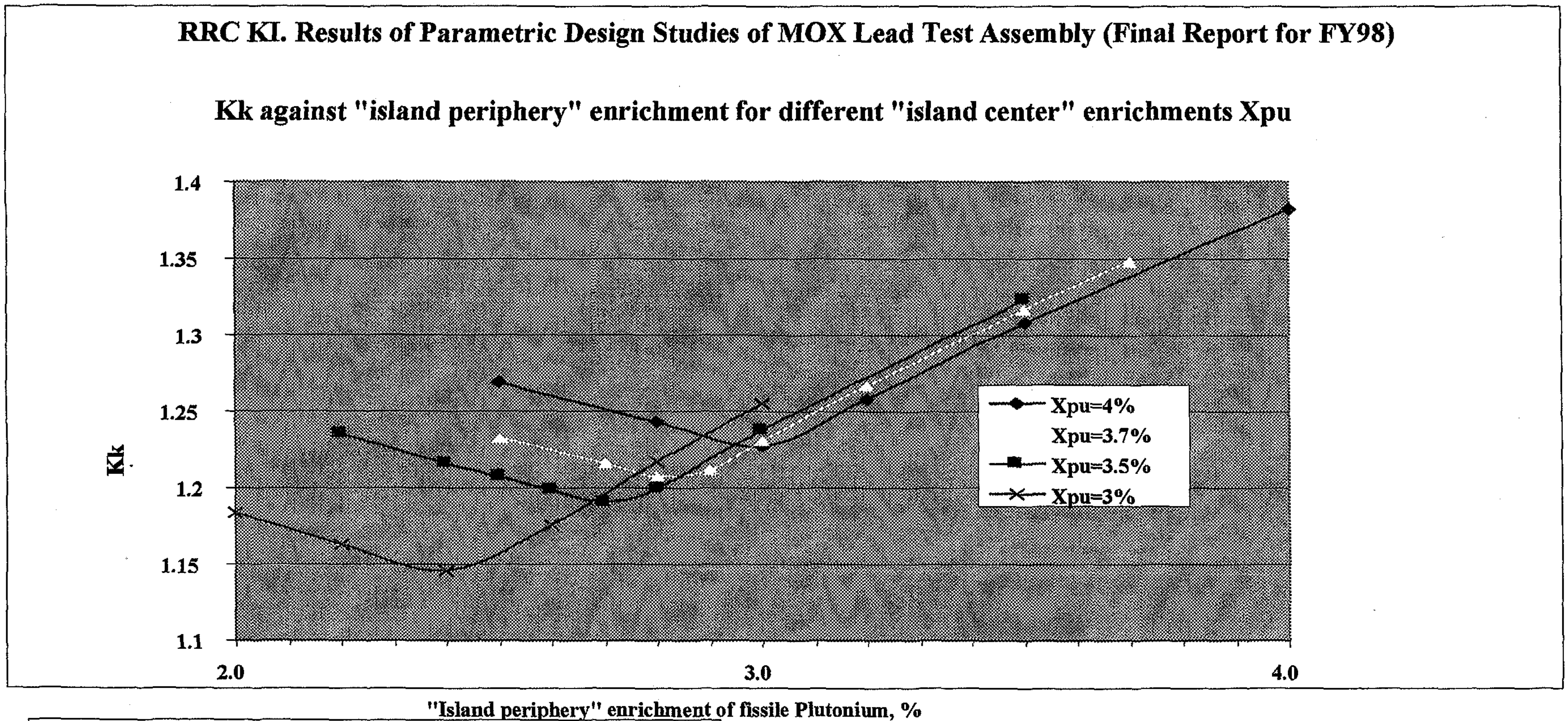
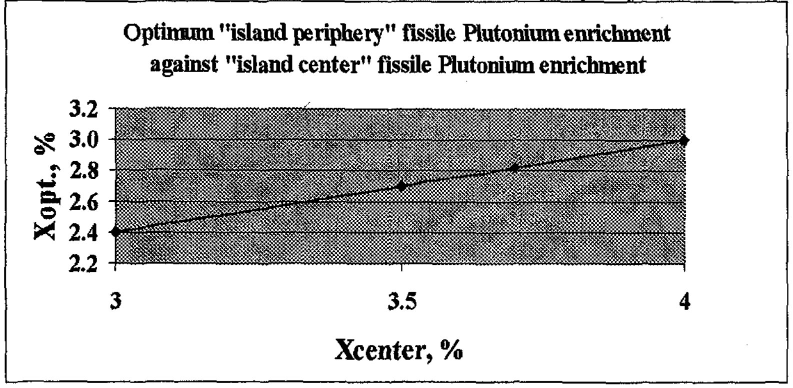


Figure 12. Parametric Studies of «Island» Type MOX LTA (U 3.7%)

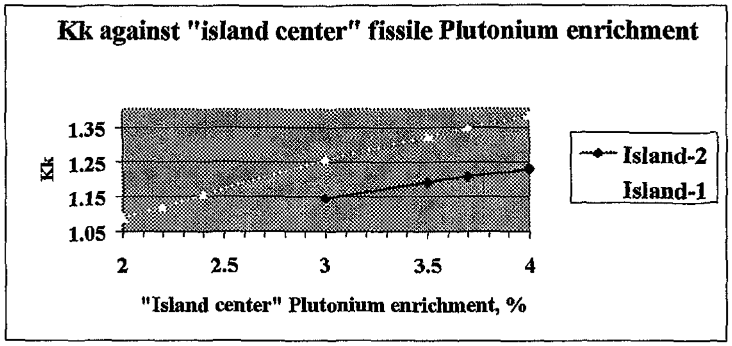


RRC KI. Results of Parametric Design Studies of MOX Lead Test Assembly (Final Report for FY98)

Kk against "island periphery" enrichment for different "island center" enrichments Xpu

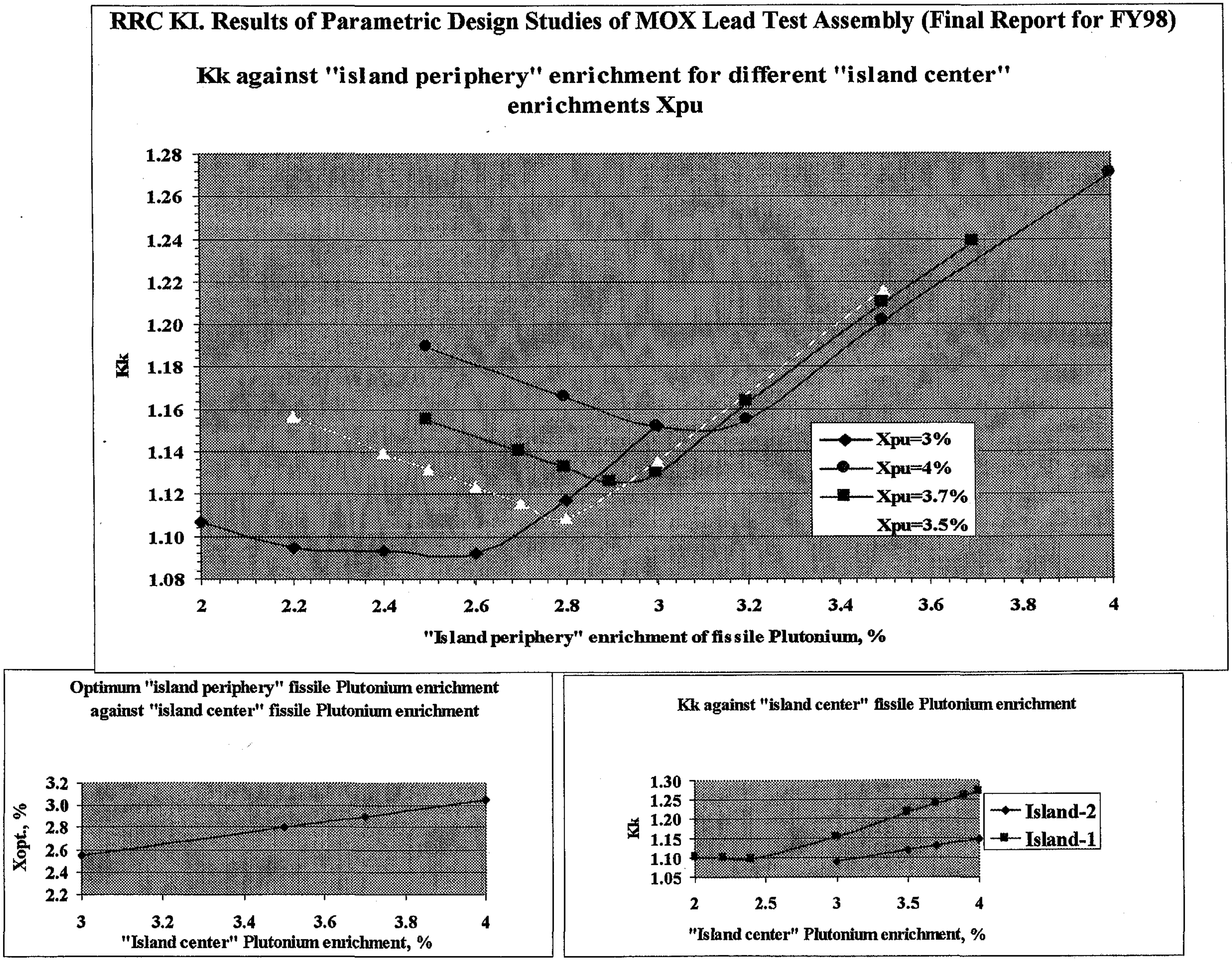


Optimum "island periphery" fissile Plutonium enrichment against "island center" fissile Plutonium enrichment



Kk against "island center" fissile Plutonium enrichment

Figure 13. Parametric Studies of «Island» Type MOX LTA (U 4.4%)



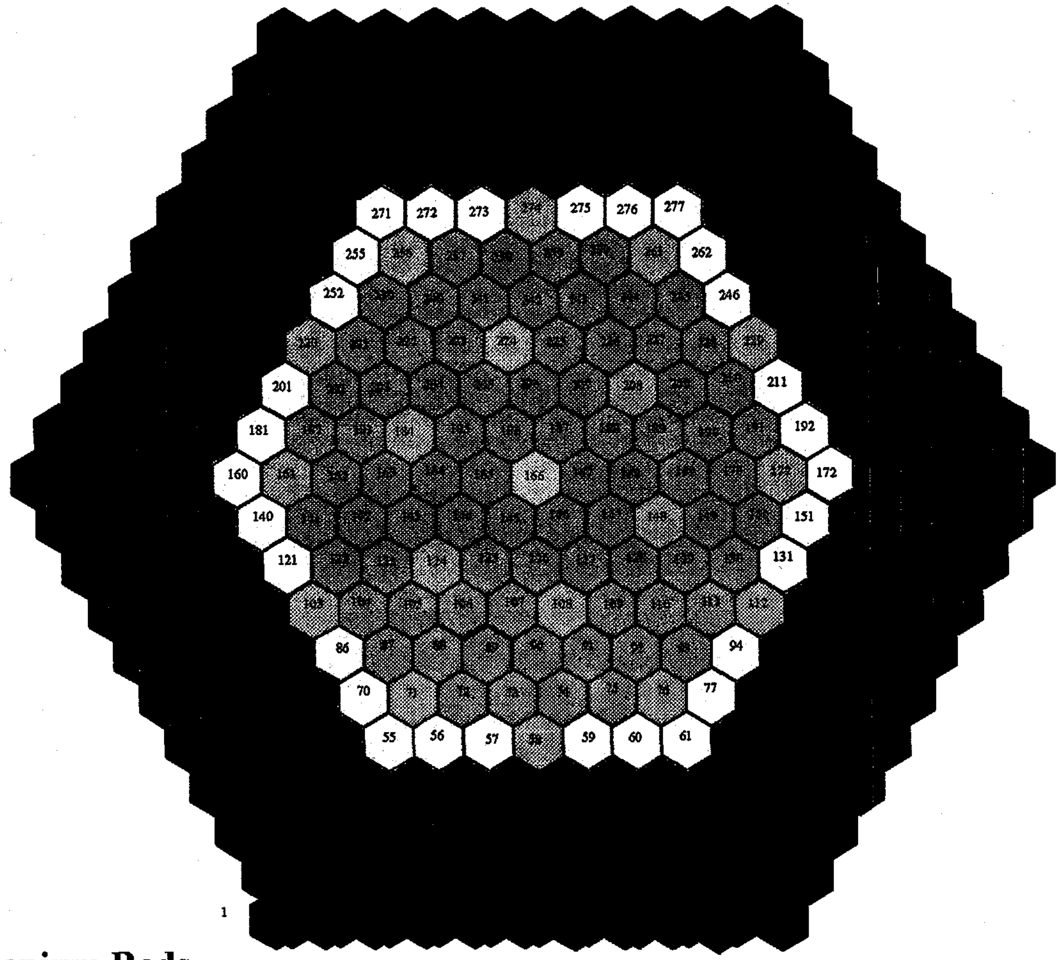
RRC KI. Results of Parametric Design Studies of MOX Lead Test Assembly (Final Report for FY98)

Kk against "island periphery" enrichment for different "island center" enrichments Xpu

Optimum "island periphery" fissile Plutonium enrichment against "island center" fissile Plutonium enrichment

Kk against "island center" fissile Plutonium enrichment

Figure 14. Simplified Design for "Increased Island-2" Type MOX LTA



⬢ **Enriched Uranium Rods**

▨ **High Plutonium-Content MOX Rods**

⬡ **Intermediate Plutonium-Content MOX Rods**

▨ **Central tube**

▨ **Control Rods / Burnable Absorbers**

Fig. 15. Kk against "Island" periphery enrichment for different "Island" size. "Island central enrichment - 4.0%. Uranium enrichment - 3.7%.

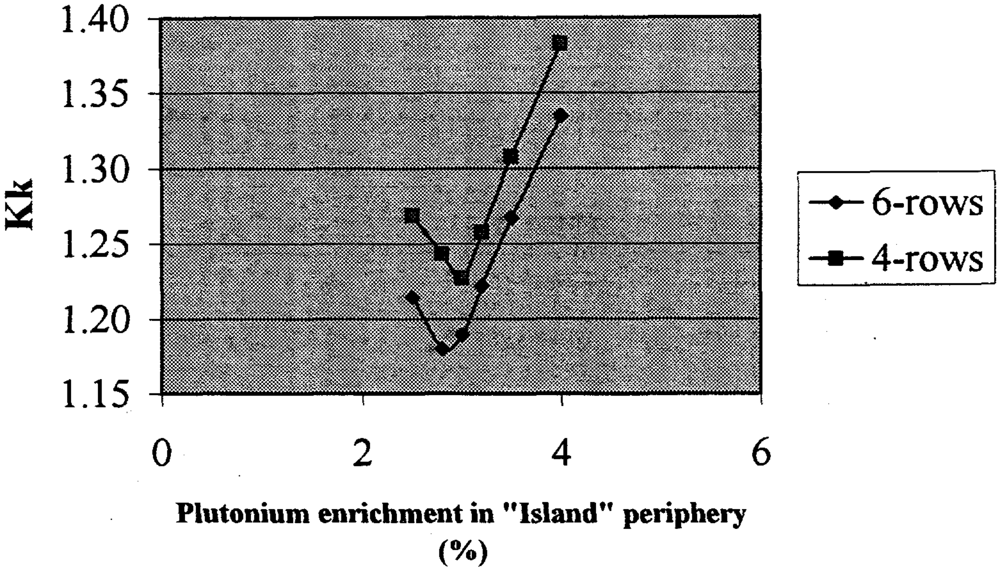
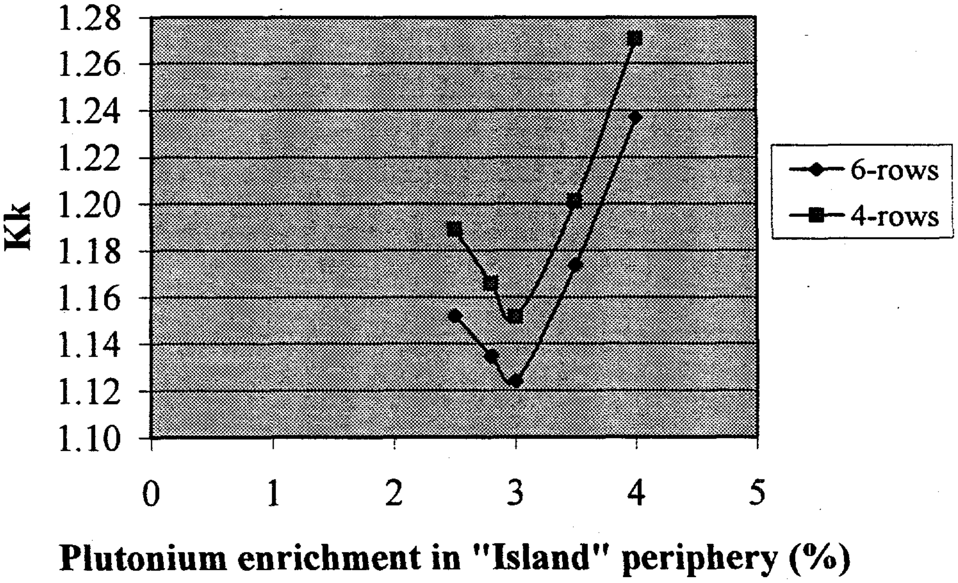


Fig. 16 Kk against "Island" periphery enrichment for different "Island" size. "Island central enrichment - 4.0%. Uranium enrichment - 4.4%.



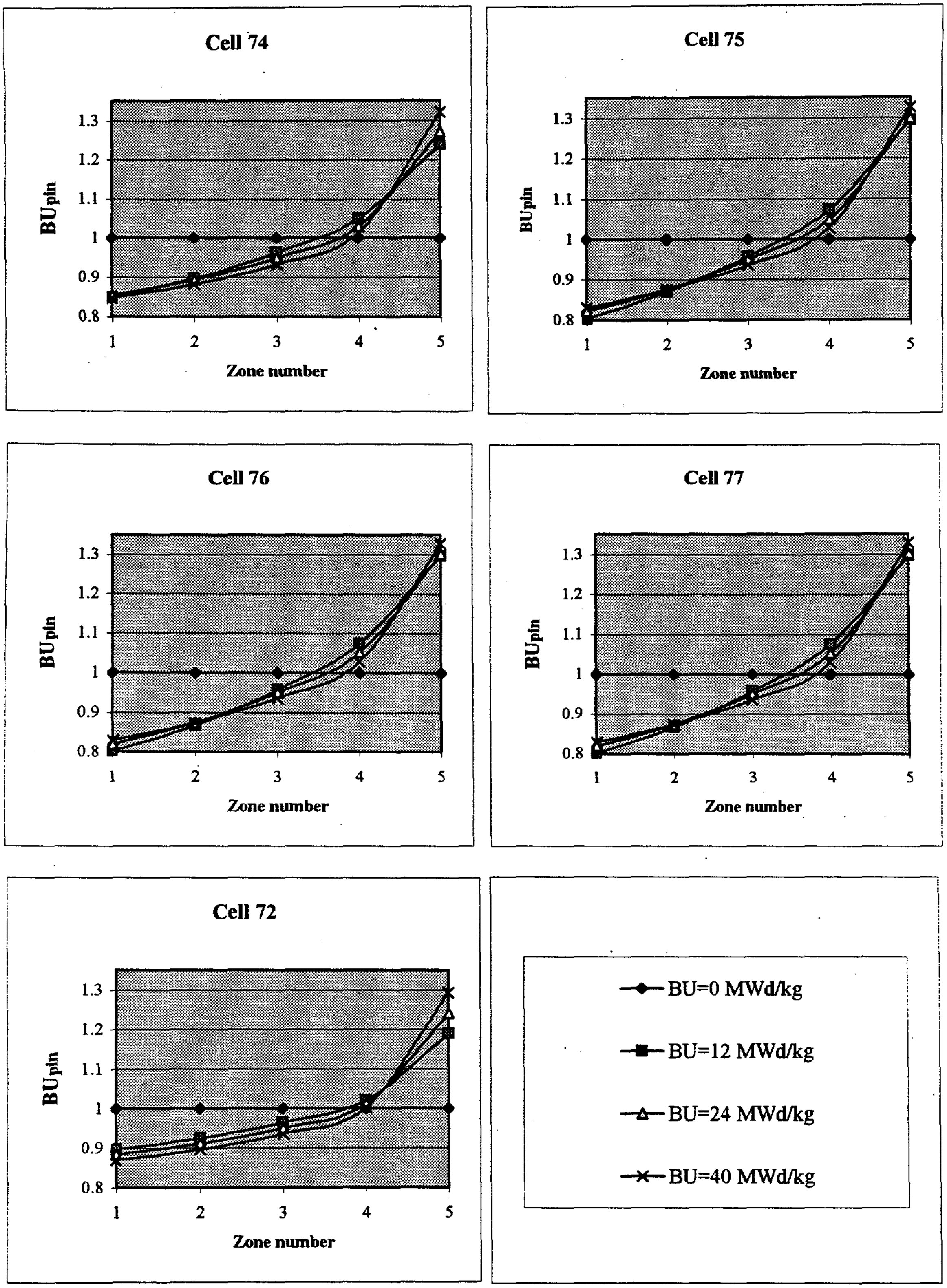


Fig. 17. Inter-pin relative burnup distribution

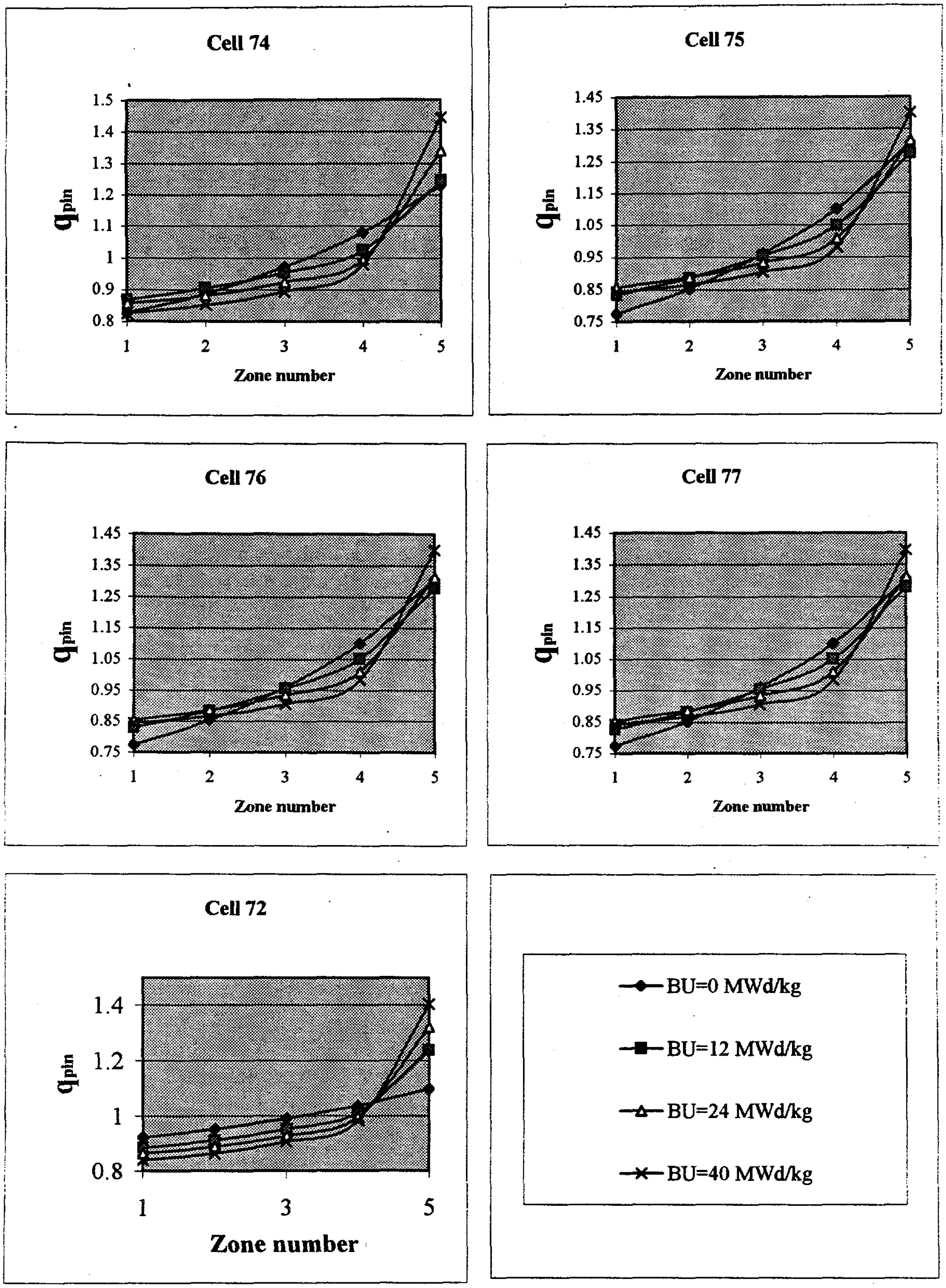


Fig. 18. Inter-pin relative power distribution

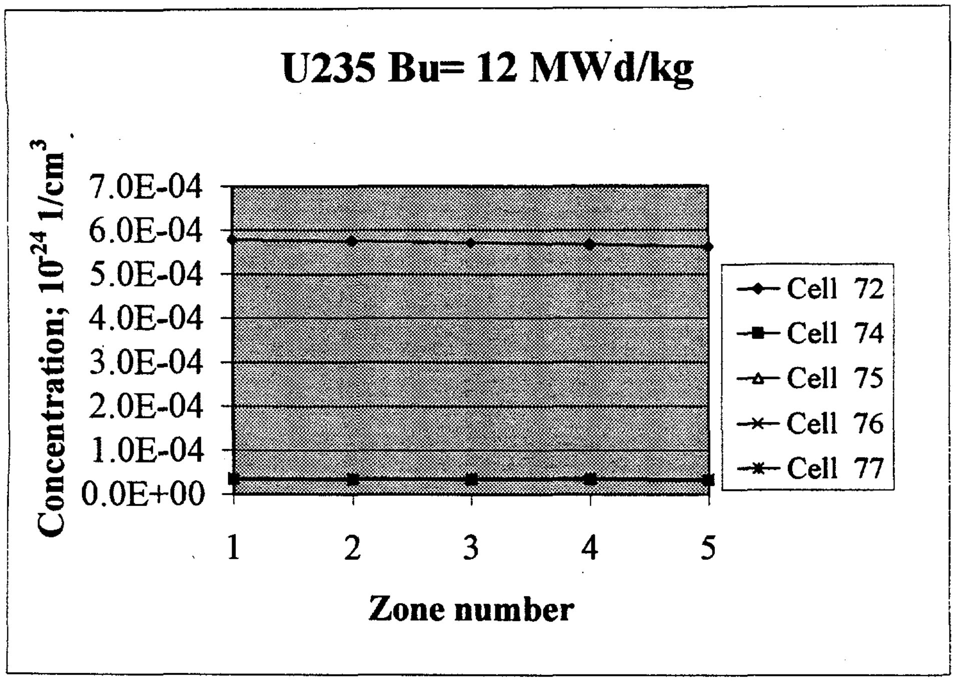


Fig. 19. Inter-pin isotopic distribution

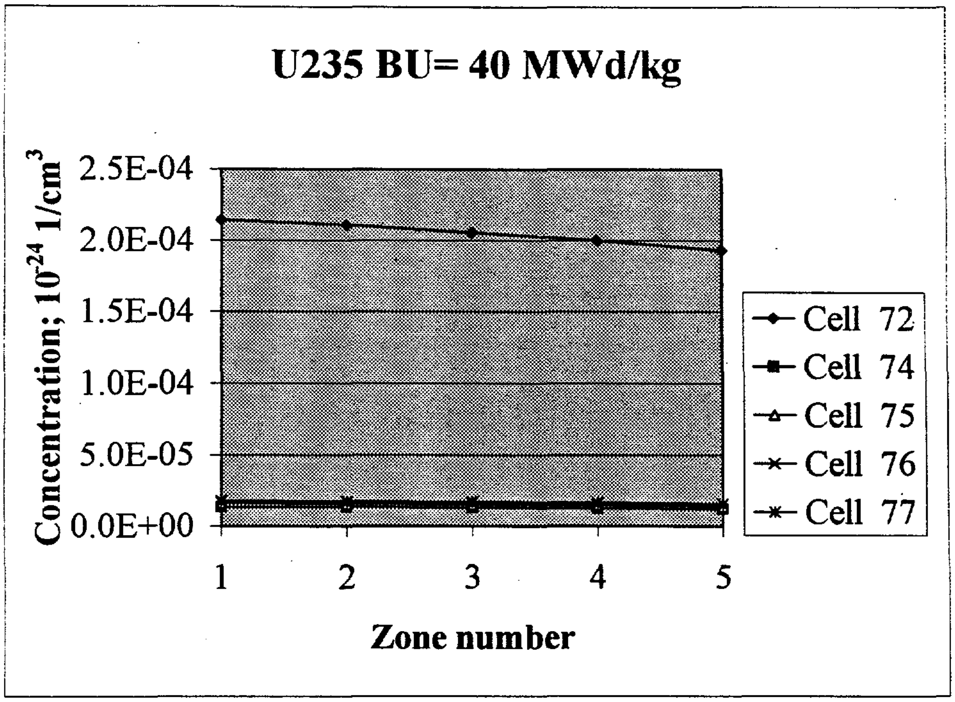


Fig. 20. Inter-pin isotopic distribution

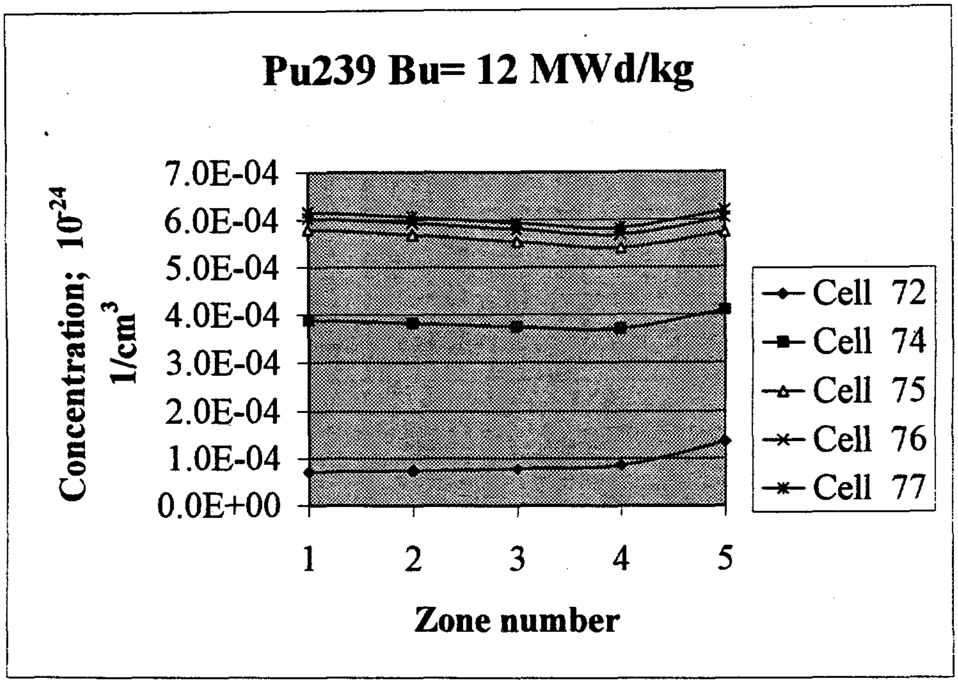


Fig. 21. Inter-pin isotopic distribution

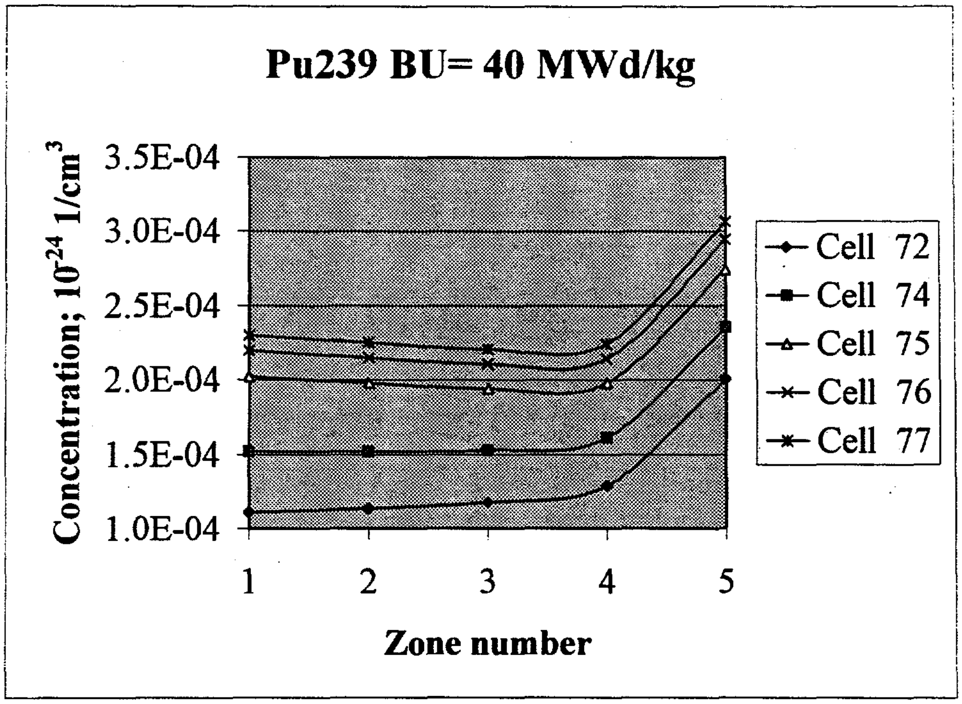


Fig. 22. Inter-pin isotopic distribution

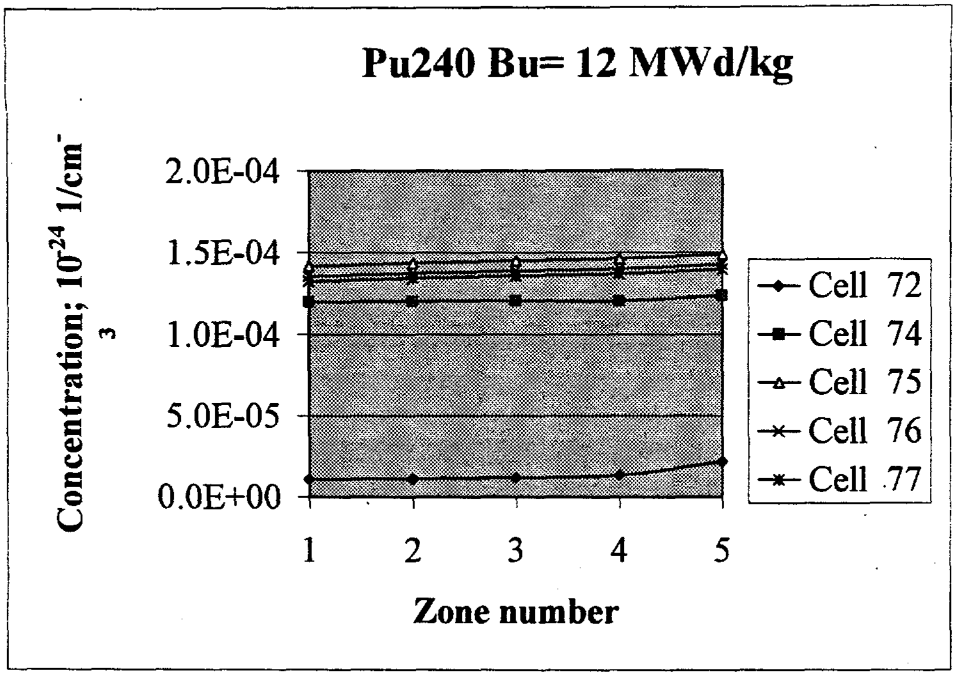


Fig. 23. Inter-pin isotopic distribution

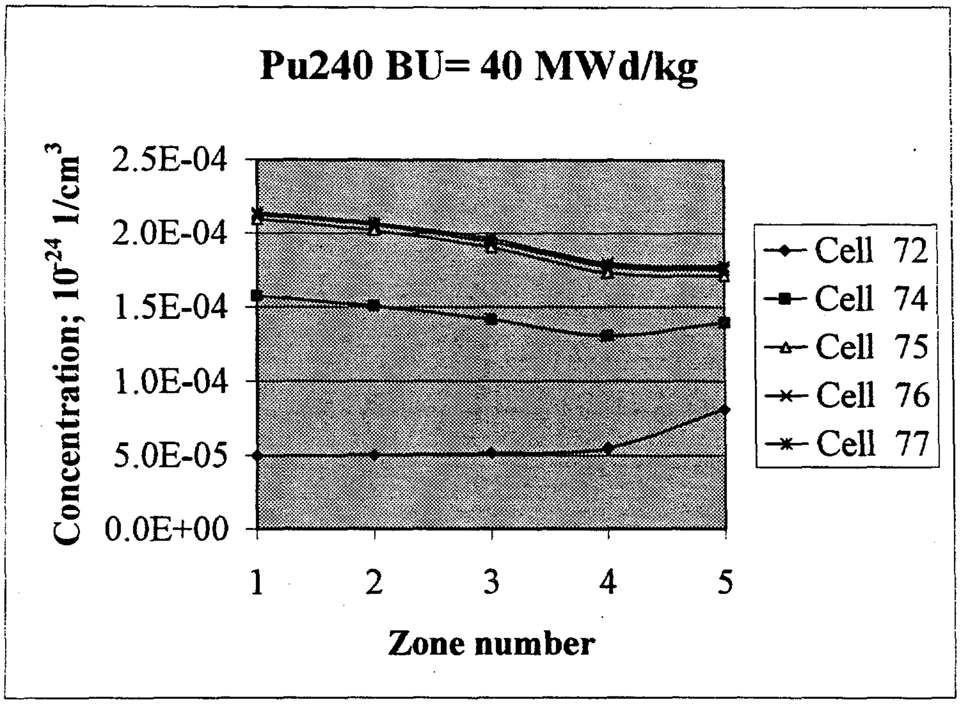


Fig. 24. Inter-pin isotopic distribution

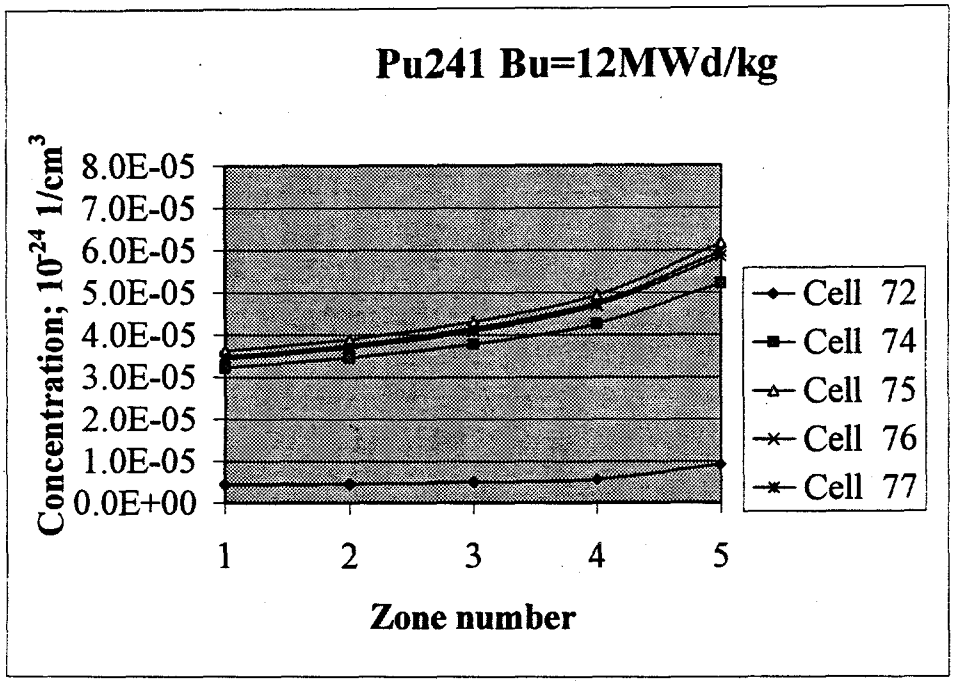


Fig. 25. Inter-pin isotopic distribution

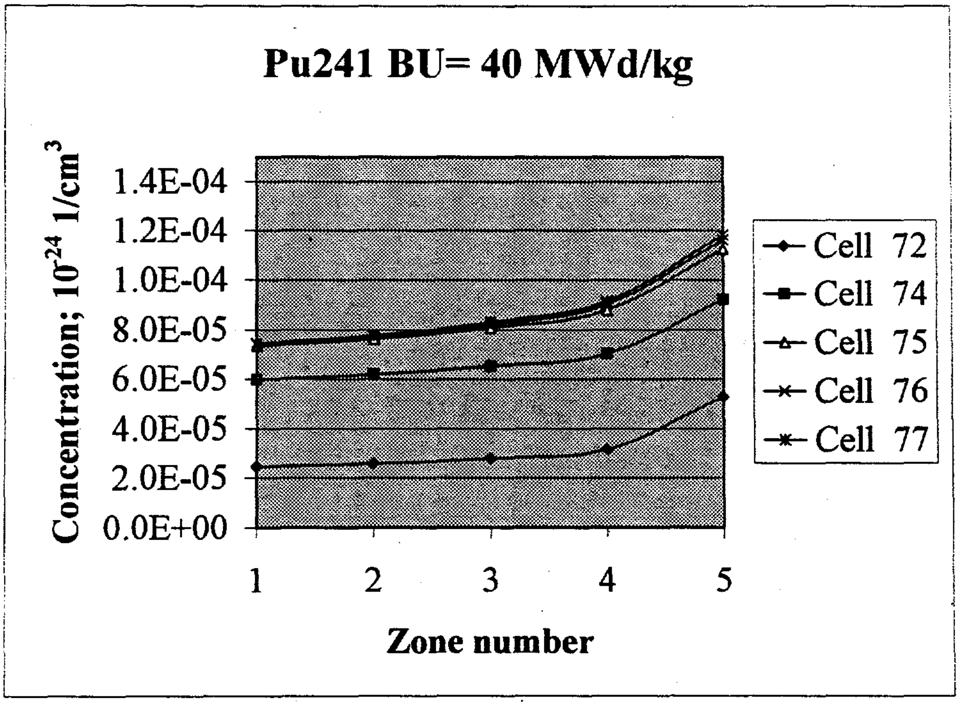


Fig. 26. Inter-pin isotopic distribution

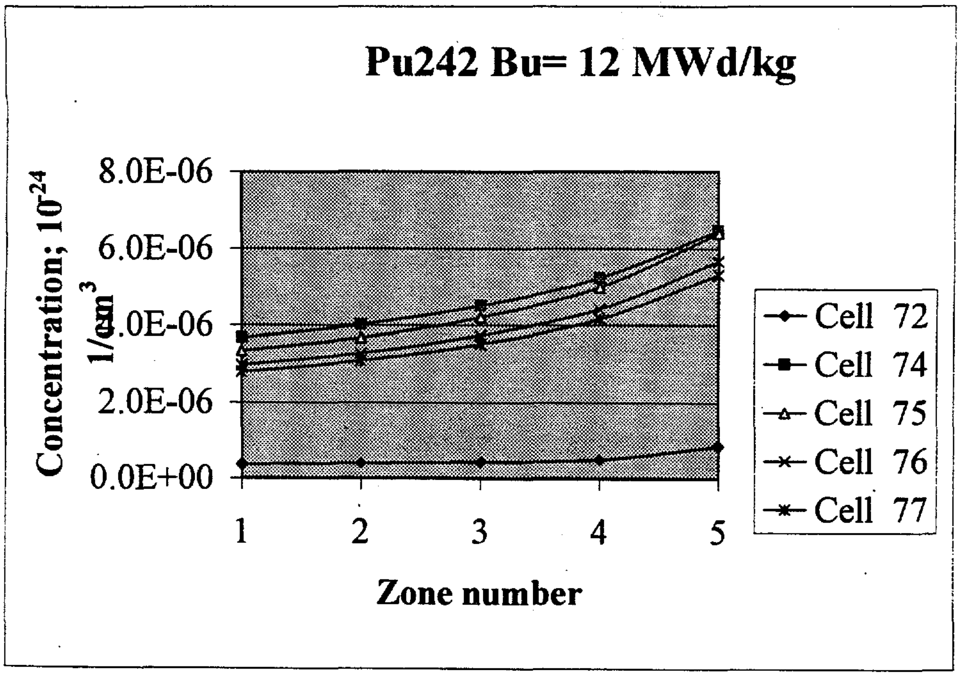


Fig. 27. Inter-pin isotopic distribution

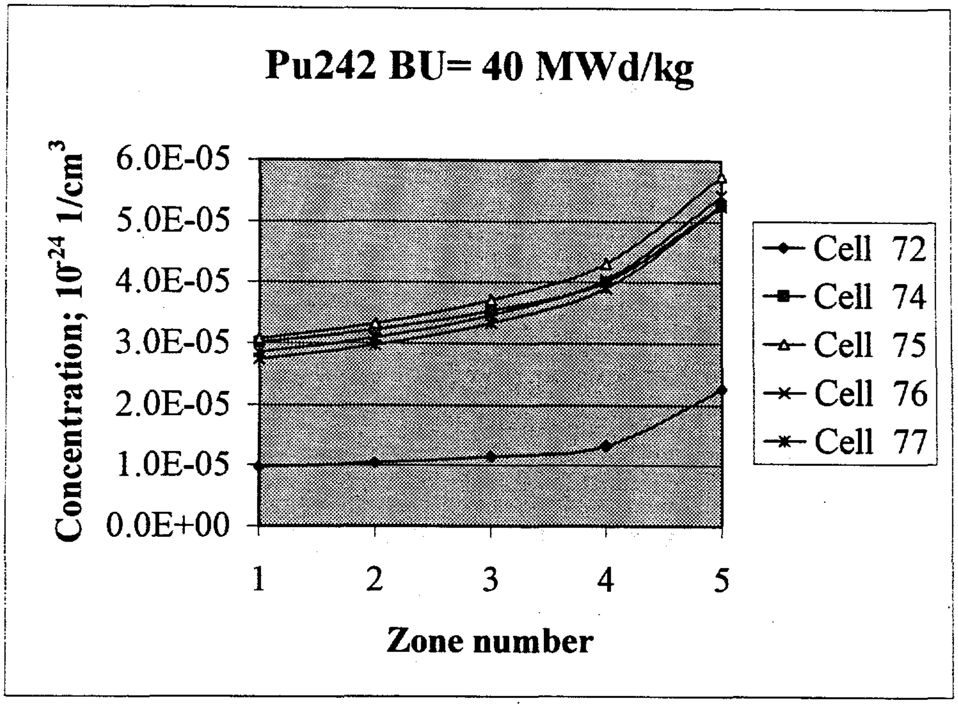


Fig. 28. Inter-pin isotopic distribution

H:\SERV AM1\DATA\pU38 pu38 .so

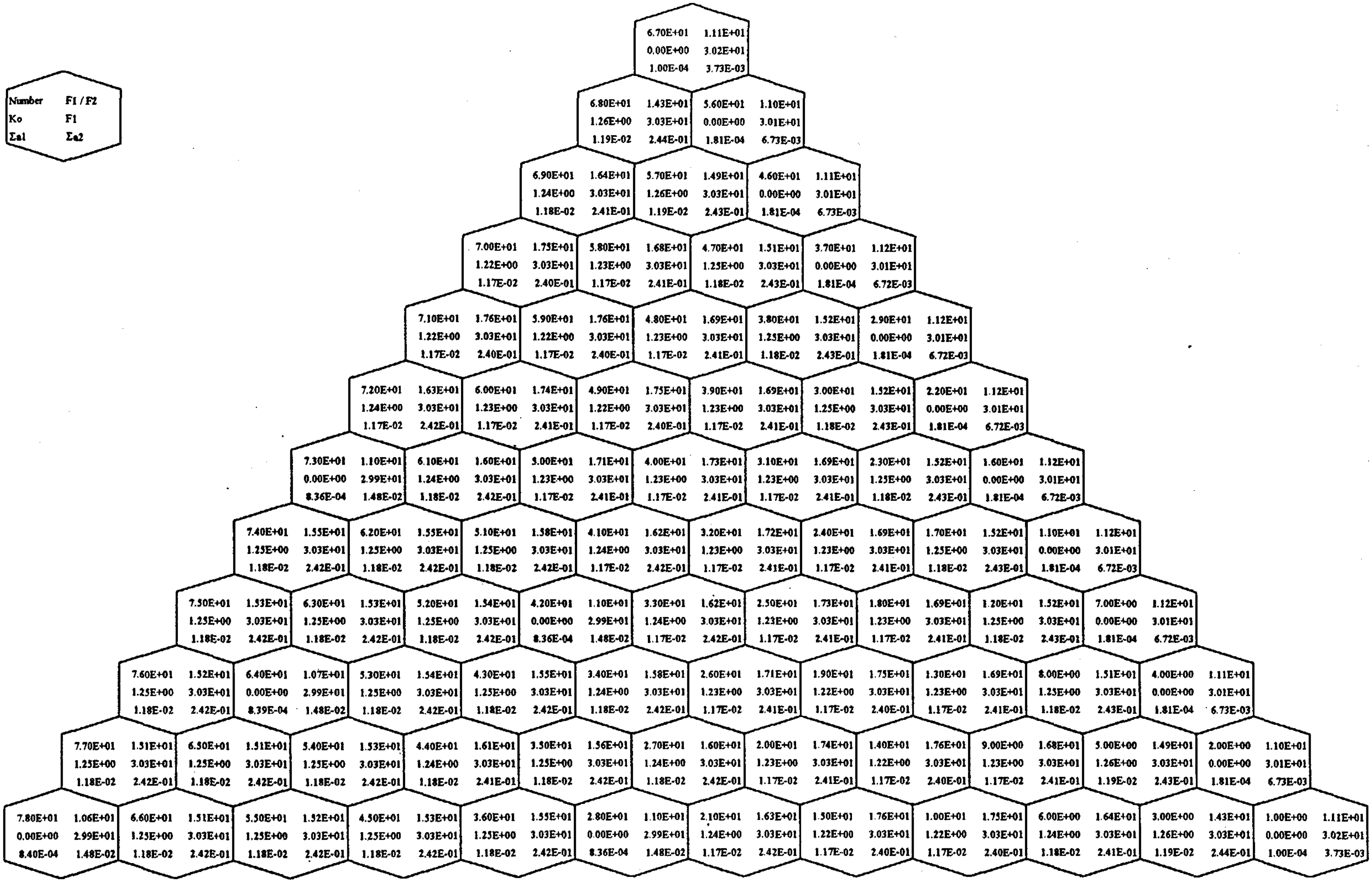
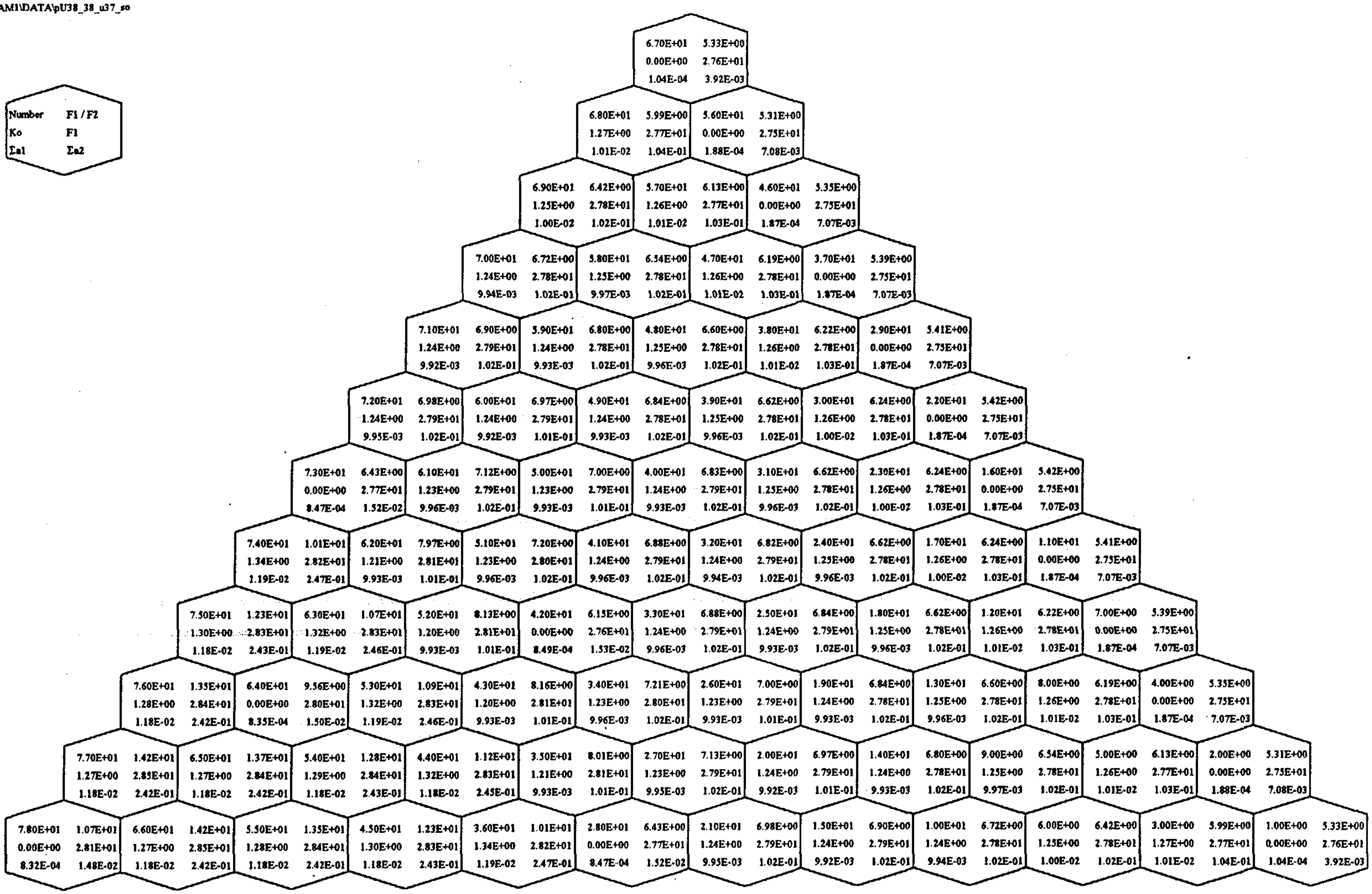


Fig. 29. Spectrum parameters distribution in MOX assembly (Pu 3.8. Sector 60°)

H:\SERV_AM1\DATA\pU38_38_u37_so



Number	F1 / F2
Ko	F1
Σa1	Σa2

6.70E+01 5.33E+00 0.00E+00 2.76E+01 1.04E-04 3.92E-03											
6.80E+01 5.99E+00 1.27E+00 2.77E+01 1.01E-02 1.04E-01	5.60E+01 5.31E+00 0.00E+00 2.75E+01 1.88E-04 7.08E-03										
6.90E+01 6.42E+00 1.25E+00 2.78E+01 1.00E-02 1.02E-01	5.70E+01 6.13E+00 1.26E+00 2.77E+01 1.01E-02 1.03E-01	4.60E+01 5.35E+00 0.00E+00 2.75E+01 1.87E-04 7.07E-03									
7.00E+01 6.72E+00 1.24E+00 2.78E+01 9.94E-03 1.02E-01	5.80E+01 6.54E+00 1.25E+00 2.78E+01 9.97E-03 1.02E-01	4.70E+01 6.19E+00 1.26E+00 2.78E+01 1.01E-02 1.03E-01	3.70E+01 5.39E+00 0.00E+00 2.75E+01 1.87E-04 7.07E-03								
7.10E+01 6.90E+00 1.24E+00 2.79E+01 9.92E-03 1.02E-01	5.90E+01 6.80E+00 1.24E+00 2.78E+01 9.93E-03 1.02E-01	4.80E+01 6.60E+00 1.25E+00 2.78E+01 9.96E-03 1.02E-01	3.80E+01 6.22E+00 1.26E+00 2.78E+01 1.01E-02 1.03E-01	2.90E+01 5.41E+00 0.00E+00 2.75E+01 1.87E-04 7.07E-03							
7.20E+01 6.98E+00 1.24E+00 2.79E+01 9.95E-03 1.02E-01	6.00E+01 6.97E+00 1.24E+00 2.79E+01 9.92E-03 1.01E-01	4.90E+01 6.84E+00 1.24E+00 2.78E+01 9.93E-03 1.02E-01	3.90E+01 6.62E+00 1.25E+00 2.78E+01 9.96E-03 1.02E-01	3.00E+01 6.24E+00 1.26E+00 2.78E+01 1.00E-02 1.03E-01	2.20E+01 5.42E+00 0.00E+00 2.75E+01 1.87E-04 7.07E-03						
7.30E+01 6.43E+00 0.00E+00 2.77E+01 8.47E-04 1.52E-02	6.10E+01 7.12E+00 1.23E+00 2.79E+01 9.96E-03 1.02E-01	5.00E+01 7.00E+00 1.23E+00 2.79E+01 9.93E-03 1.01E-01	4.00E+01 6.83E+00 1.24E+00 2.79E+01 9.93E-03 1.02E-01	3.10E+01 6.62E+00 1.25E+00 2.78E+01 9.96E-03 1.02E-01	2.30E+01 6.24E+00 1.26E+00 2.78E+01 1.00E-02 1.03E-01	1.60E+01 5.42E+00 0.00E+00 2.75E+01 1.87E-04 7.07E-03					
7.40E+01 1.01E+01 1.34E+00 2.82E+01 1.19E-02 2.47E-01	6.20E+01 7.97E+00 1.21E+00 2.81E+01 9.93E-03 1.01E-01	5.10E+01 7.20E+00 1.23E+00 2.80E+01 9.96E-03 1.02E-01	4.10E+01 6.88E+00 1.24E+00 2.79E+01 9.96E-03 1.02E-01	3.20E+01 6.82E+00 1.24E+00 2.79E+01 9.94E-03 1.02E-01	2.40E+01 6.62E+00 1.25E+00 2.78E+01 9.96E-03 1.02E-01	1.70E+01 6.24E+00 1.26E+00 2.78E+01 1.00E-02 1.03E-01	1.10E+01 5.41E+00 0.00E+00 2.75E+01 1.87E-04 7.07E-03				
7.50E+01 1.23E+01 1.30E+00 2.83E+01 1.18E-02 2.43E-01	6.30E+01 1.07E+01 1.32E+00 2.83E+01 1.19E-02 2.46E-01	5.20E+01 8.13E+00 1.20E+00 2.81E+01 9.93E-03 1.01E-01	4.20E+01 6.15E+00 0.00E+00 2.76E+01 8.49E-04 1.53E-02	3.30E+01 6.88E+00 1.24E+00 2.79E+01 9.96E-03 1.02E-01	2.50E+01 6.84E+00 1.24E+00 2.79E+01 9.93E-03 1.02E-01	1.80E+01 6.62E+00 1.25E+00 2.78E+01 9.96E-03 1.02E-01	1.20E+01 6.22E+00 1.26E+00 2.78E+01 1.01E-02 1.03E-01	7.00E+00 5.39E+00 0.00E+00 2.75E+01 1.87E-04 7.07E-03			
7.60E+01 1.35E+01 1.28E+00 2.84E+01 1.18E-02 2.42E-01	6.40E+01 9.56E+00 0.00E+00 2.80E+01 8.35E-04 1.50E-02	5.30E+01 1.09E+01 1.32E+00 2.83E+01 1.19E-02 2.46E-01	4.30E+01 8.16E+00 1.20E+00 2.81E+01 9.93E-03 1.01E-01	3.40E+01 7.21E+00 1.23E+00 2.80E+01 9.96E-03 1.02E-01	2.60E+01 7.00E+00 1.23E+00 2.79E+01 9.93E-03 1.01E-01	1.90E+01 6.84E+00 1.24E+00 2.78E+01 9.93E-03 1.02E-01	1.30E+01 6.60E+00 1.25E+00 2.78E+01 9.96E-03 1.02E-01	8.00E+00 6.19E+00 1.26E+00 2.78E+01 1.01E-02 1.03E-01	4.00E+00 5.35E+00 0.00E+00 2.75E+01 1.87E-04 7.07E-03		
7.70E+01 1.42E+01 1.27E+00 2.85E+01 1.18E-02 2.42E-01	6.50E+01 1.37E+01 1.27E+00 2.84E+01 1.18E-02 2.42E-01	5.40E+01 1.28E+01 1.29E+00 2.84E+01 1.18E-02 2.43E-01	4.40E+01 1.12E+01 1.32E+00 2.83E+01 1.18E-02 2.45E-01	3.50E+01 8.01E+00 1.21E+00 2.81E+01 9.93E-03 1.01E-01	2.70E+01 7.13E+00 1.23E+00 2.79E+01 9.95E-03 1.02E-01	2.00E+01 6.97E+00 1.24E+00 2.79E+01 9.92E-03 1.01E-01	1.40E+01 6.80E+00 1.24E+00 2.78E+01 9.93E-03 1.02E-01	9.00E+00 6.54E+00 1.25E+00 2.78E+01 9.97E-03 1.02E-01	5.00E+00 6.13E+00 1.26E+00 2.77E+01 1.01E-02 1.03E-01	2.00E+00 5.31E+00 0.00E+00 2.75E+01 1.88E-04 7.08E-03	
7.80E+01 1.07E+01 0.00E+00 2.81E+01 8.32E-04 1.48E-02	6.60E+01 1.42E+01 1.27E+00 2.85E+01 1.18E-02 2.42E-01	5.50E+01 1.35E+01 1.28E+00 2.84E+01 1.18E-02 2.42E-01	4.50E+01 1.23E+01 1.30E+00 2.83E+01 1.18E-02 2.43E-01	3.60E+01 1.01E+01 1.34E+00 2.82E+01 1.19E-02 2.47E-01	2.80E+01 6.43E+00 0.00E+00 2.77E+01 8.47E-04 1.52E-02	2.10E+01 6.98E+00 1.24E+00 2.79E+01 9.95E-03 1.02E-01	1.50E+01 6.90E+00 1.24E+00 2.79E+01 9.92E-03 1.02E-01	1.00E+01 6.72E+00 1.24E+00 2.78E+01 9.94E-03 1.02E-01	6.00E+00 6.42E+00 1.25E+00 2.78E+01 1.00E-02 1.02E-01	3.00E+00 5.99E+00 1.27E+00 2.77E+01 1.01E-02 1.04E-01	1.00E+00 5.33E+00 0.00E+00 2.76E+01 1.04E-04 3.92E-03

Fig. 30. Spectrum parameters distribution in "Island" type MOX assembly (Pu 3.8_3.8_U 3.7. Sector 60°)

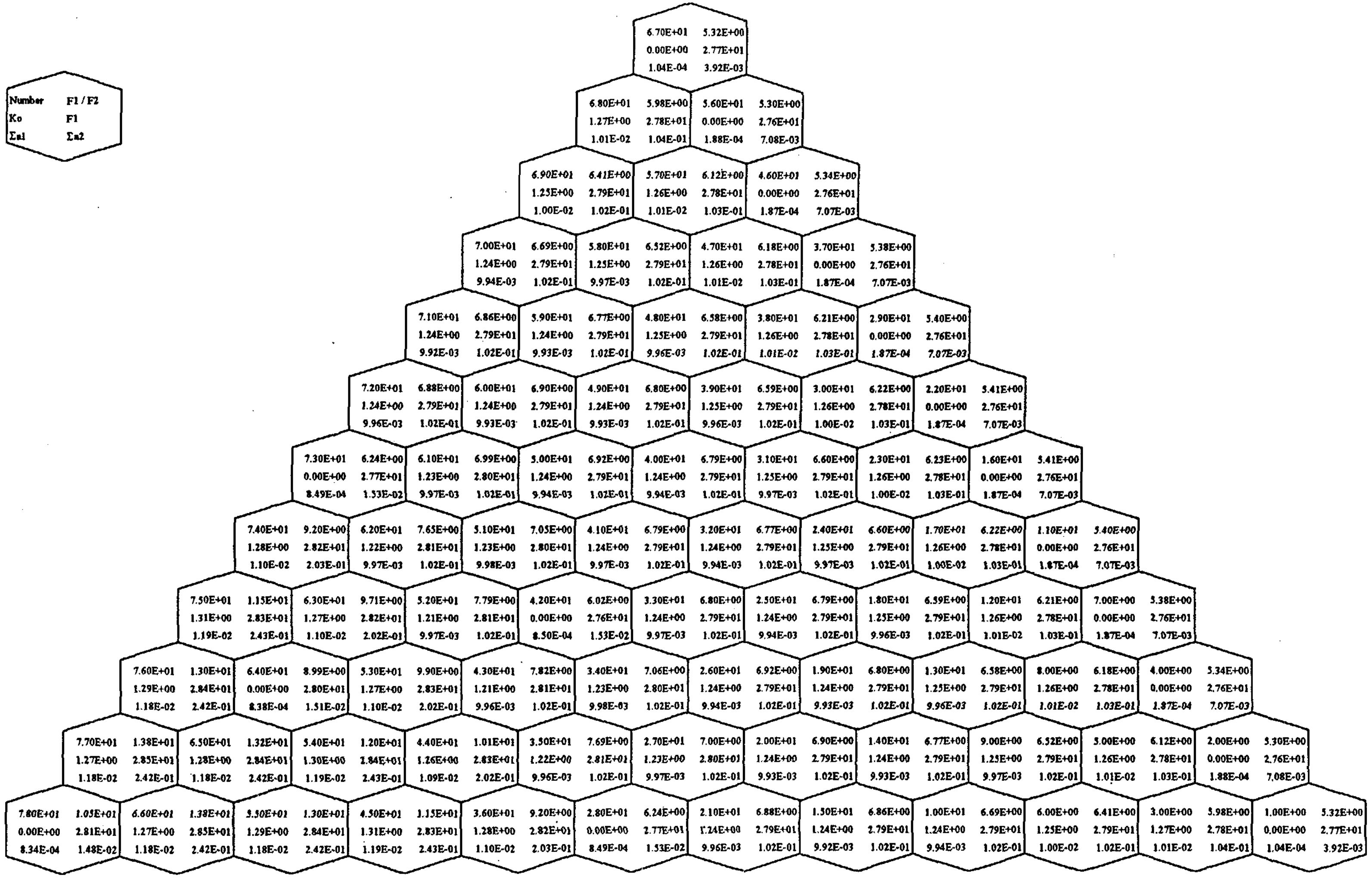


Fig. 31. Spectrum parameters distribution in "Island" type MOX assembly (Pu 3.8_2.8_U 3.7. Sector 60o)

pu38_28_u37o
Current Burnup 24 MWtd/kg
Power Distribution

										68
										1.034
									69	57
									1.003	1.024
								70	58	47
								0.981	0.993	1.019
							71	59	48	38
							0.973	0.976	0.99	1.016
						72	60	49	39	30
						0.986	0.974	0.976	0.989	1.015
					73	61	50	40	31	23
					0	0.989	0.977	0.978	0.989	1.015
				74	62	51	41	32	24	17
				0.951	0.991	0.991	0.988	0.98	0.989	1.015
			75	63	52	42	33	25	18	12
			1.108	0.954	0.99	0	0.988	0.978	0.989	1.016
		76	64	53	43	34	26	19	13	8
		1.1	0	0.953	0.988	0.99	0.977	0.976	0.99	1.019
	77	65	54	44	35	27	20	14	9	5
	1.091	1.097	1.105	0.948	0.989	0.988	0.974	0.976	0.993	1.024
78	66	55	45	36	28	21	15	10	6	3
0	1.091	1.1	1.108	0.951	0	0.986	0.973	0.981	1.003	1.034

pu38_28_u37o
Current Burnup 40 MWtd/kg
Power Distribution

										68
										1.011
									69	57
									0.998	1.006
								70	58	47
								0.987	0.993	1.004
							71	59	48	38
							0.984	0.985	0.991	1.003
						72	60	49	39	30
						0.994	0.986	0.985	0.991	1.002
					73	61	50	40	31	23
					0	0.999	0.989	0.987	0.991	1.002
				74	62	51	41	32	24	17
				0.942	1.014	1.003	0.995	0.988	0.991	1.002
			75	63	52	42	33	25	18	12
			1.061	0.949	1.016	0	0.995	0.987	0.991	1.003
		76	64	53	43	34	26	19	13	8
		1.078	0	0.95	1.016	1.002	0.989	0.985	0.991	1.004
	77	65	54	44	35	27	20	14	9	5
	1.086	1.081	1.067	0.947	1.013	0.999	0.986	0.985	0.993	1.006
78	66	55	45	36	28	21	15	10	6	3
0	1.086	1.078	1.061	0.942	0	0.994	0.984	0.987	0.998	1.011

Fig. 33. Power distribution evolution in "Island" type MOX assembly (Pu3.8_2.8_U3.7 Sector 60°)

pu38_28_u37o
Current Burnup 12 MWtd/kg
Burnup Distribution (MWtd/kg)

										68 12.854
									69 12.113	57 12.603
								70 11.693	58 11.935	47 12.501
							71 11.498	59 11.596	48 11.863	38 12.451
						72 11.575	60 11.465	49 11.566	39 11.84	30 12.43
					73 0	61 11.498	50 11.468	40 11.588	31 11.84	23 12.425
				74 13.065	62 10.978	51 11.461	41 11.67	32 11.614	24 11.84	17 12.43
			75 14.05	63 12.696	52 10.871	42 0	33 11.667	25 11.587	18 11.84	12 12.45
		76 13.067	64 0	53 12.556	43 10.832	34 11.446	26 11.463	19 11.564	13 11.862	8 12.5
	77 12.58	65 12.914	54 13.663	44 12.362	35 10.932	27 11.488	20 11.463	14 11.595	9 11.934	5 12.603
78 0	66 12.58	55 13.067	45 14.05	36 13.065	28 0	21 11.575	15 11.498	10 11.693	6 12.113	3 12.854

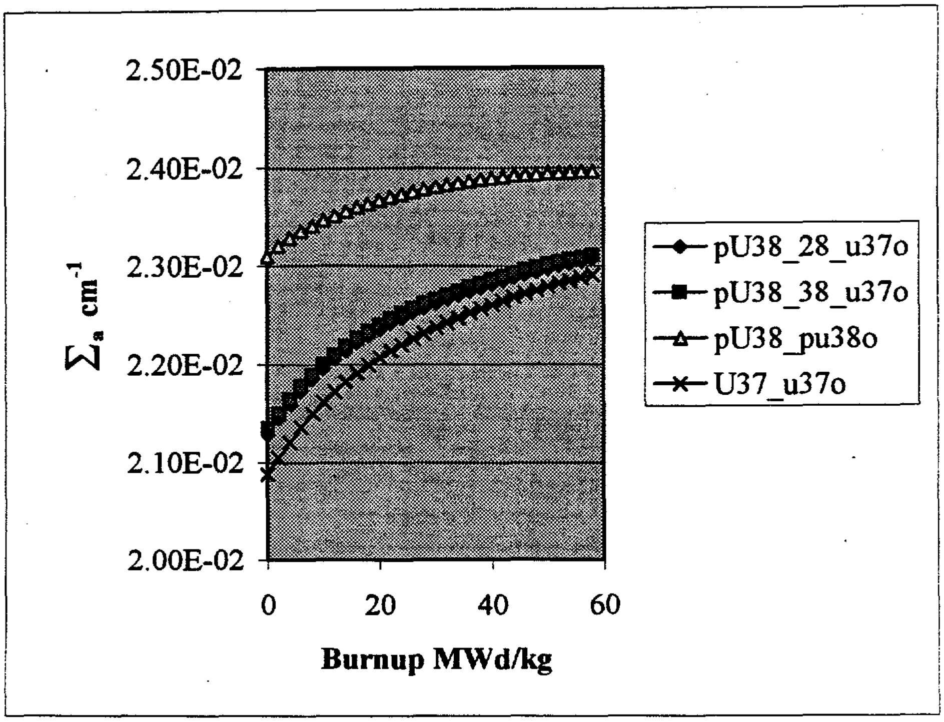
pu38_28_u37o
Current Burnup 24 MWtd/kg
Burnup Distribution (MWtd/kg)

										68 25.456
									69 24.2	57 25.032
								70 23.457	58 23.884	47 24.852
							71 23.135	59 23.291	48 23.758	38 24.763
						72 23.358	60 23.106	49 23.248	39 23.72	30 24.727
					73 0	61 23.289	50 23.138	40 23.298	31 23.722	23 24.718
				74 24.718	62 22.645	51 23.264	41 23.506	32 23.348	24 23.722	17 24.727
			75 27.477	63 24.309	52 22.495	42 0	33 23.501	25 23.295	18 23.719	12 24.763
		76 26.134	64 0	53 24.123	43 22.423	34 23.237	26 23.128	19 23.244	13 23.756	8 24.851
	77 25.411	65 25.906	54 26.955	44 23.832	35 22.562	27 23.27	20 23.101	14 23.289	9 23.884	5 25.032
78 0	66 25.411	55 26.134	45 27.477	36 24.718	28 0	21 23.358	15 23.135	10 23.457	6 24.2	3 25.456

pu38_28_u37o
Current Burnup 40 MWtd/kg
Burnup Distribution (MWtd/kg)

										68 41.878
									69 40.256	57 41.334
								70 39.245	58 39.826	47 41.089
							71 38.834	59 39.024	48 39.653	38 40.968
						72 39.248	60 38.832	49 38.979	39 39.605	30 40.919
					73 0	61 39.244	50 38.912	40 39.061	31 39.61	23 40.907
				74 39.568	62 38.752	51 39.268	41 39.423	32 39.134	24 39.609	17 40.918
			75 44.589	63 39.246	52 38.612	42 0	33 39.416	25 39.056	18 39.602	12 40.967
		76 43.342	64 0	53 39.06	43 38.515	34 39.23	26 38.899	19 38.974	13 39.651	8 41.088
	77 42.616	65 43.112	54 44.107	44 38.71	35 38.64	27 39.218	20 38.825	14 39.021	9 39.825	5 41.333
78 0	66 42.616	55 43.342	45 44.589	36 39.568	28 0	21 39.248	15 38.834	10 39.245	6 40.256	3 41.878

Fig. 34. Burnup distribution evolution in "Island" type MOX assembly (Pu3.8 2.8 U3.7 Sector 60^{0})



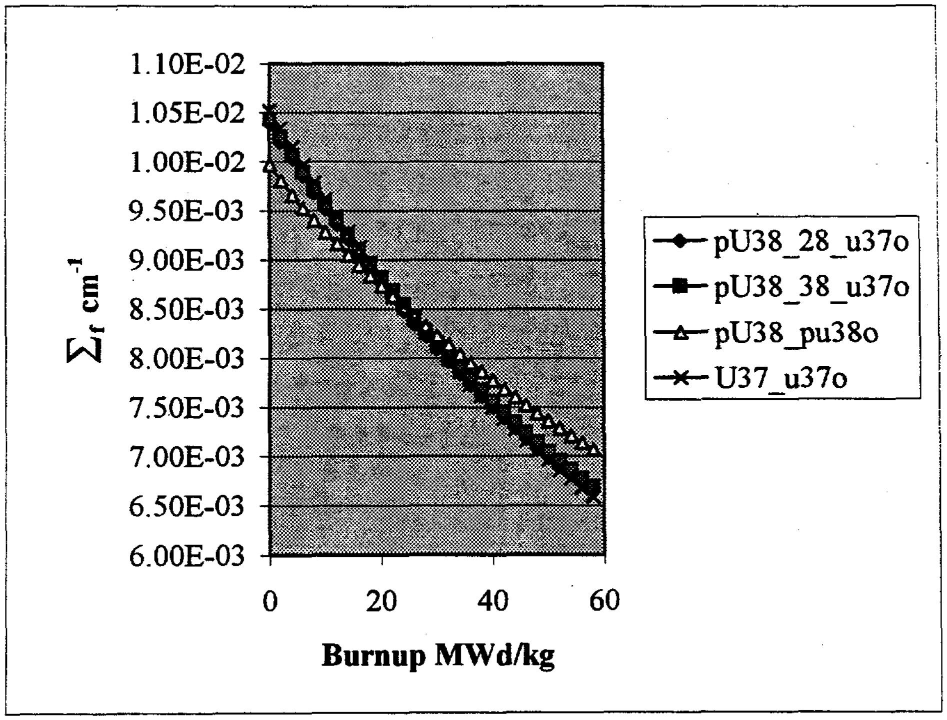


Fig. 35. Assembly parameters evolution for different enrichment compositions