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Design Studies of “Island” Type MOX Lead Test Assembly

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Fissile Materials Disposition Program

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Design Studies of «Island» Type MOX Lead Test Assembly

(Final Report for FY99)

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ACRONYMS

Russian		American Equivalent
AZ	emergency (accident) protection	AP
AZ-1	state with all the control rods fully inserted except of one the most effective stuck in upper position	AP-1
BOC	Beginning Of fuel Cycle	BOC
BPR	Burnable Poison Rod	BPR
DNBR	Departure from Nucleate Boiling Ratio	DNBR
DTC	Doppler Temperature Coefficient	DTC
EFPD	Effective Full Power Day	EFPD
EOC	End Of fuel Cycle	EOC
FP	Fission Products	FP
KI	Kurchatov Institute	KI
LTA	Lead Test Assembly	LTA
LWR	Light Water Reactor	LWR
MCL	Minimum Controllable reactor power Level	MCL
MDC	Moderator Density Coefficient	MDC
MOX	Mixed Oxide (uranium-plutonium fuel)	MOX
MTC	Moderator Temperature Coefficient	MTC
NPP	Nuclear Power Plant	NPP
OR	Regulatory Body (Control Rod)	CR
PWR	Pressurized-Water Reactor	PWR
RCT	Repeat Criticality Temperature	RCT
SUZ	Reactor Control and Protection System	RPS
TVS, FA	Fuel Assembly	FA
UOX	Uranium Oxide Fuel	UOX
VVER	Russian water-water reactor	VVER

EXECUTIVE SUMMARY

In this document the results of neutronics studies of «Island» type MOX LTA design are presented. The characteristics both for infinite MOX grids and for VVER-1000 core with 3 MOX LTAs are calculated. The neutronics parameters of MOX fuelled core have been performed using the Russian 3D code BIPR-7A and 2D code PERMAK-A with the constants prepared by the cell spectrum 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 of studies of MOX LTA design of «Island» type.

Two options of «Island» are considered:

- **"Island-2"** with two regions of different plutonium enrichment, Fig.2.9 (the main case);
- **"Island-1"** with homogeneous plutonium region, Fig.2.7.

The "Island" type of MOX assembly should be studied additionally to the world-wide full scale (100% Plutonium, Fig.2.5) MOX assembly because it possesses the following advantages in comparison with 100% MOX assembly:

- two types of plutonium fuel pins instead of three,
- only uranium fuel pins, whose properties are well studied, are placed near water gap,
- low enrichment plutonium pins, not effective for plutonium burnout, are absent,
- external uranium row can be regarded as a sort of shielding for MOX assembly. It should be taken into account that no additional transport expenses will be incurred if MOX assemblies and uranium assemblies fabrication are not separated.

Besides the Plutonium region in the proposed "Island" configuration possesses the neutron spectrum close to the one in 100% Plutonium MOX LTA. It can be concluded that if MOX fuel pin fabrication for pilot irradiation in VVER-1000 is limited for any reason, "Island" type MOX LTAs can be used with the same "scientific efficiency" as 100% PU MOX LTAs.

The presented studies include the ones defined in [2] as the **stages "Assembly"** and **"Core"**. This report completes the studies partially executed in [3] and [6] and can be considered as a one compiled the previous studies of «Island» MOX LTAs and VVER-1000 core configurations with 3 MOX LTAs .

At the **stage "Assembly"** in the process of parametric studies two options of infinite grid are considered:

- grid consisting of single MOX LTAs;
- grid consisting of multi-assemblies: a central MOX LTA 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 some replaced uranium assembly that was used as a base or reference FA (Fig.2.1);

- MOX LTA zoning that ensures an acceptable power peaking factor in calculational system.

The Russian cell code TVS-M [3] is used as a calculational instrument at the stage "Assembly".

The **stage "Core"** comprises studies of characteristics of some base Uranium core (Fig.A.1) with 3 MOX LTAs introduced.

The code TVS-M is used here for generation of neutronics constants to be used in:

- coarse-mesh (assembly-by-assembly) core calculations by the Russian code BIPR-7A [7];
- fine-mesh (pin-by-pin) calculations by the Russian code PERMAK-A [7].

The stages "Assembly" and "Core" are described correspondingly in Chapters 2 and 3.

In Chapter 2 additionally to [3] the studies on stability of optimal zoning (i.e. with minimal power peaking factor) are described, particularly, influence of boron concentration in coolant.

In Annex the used codes are briefly described and the detailed reflector description is presented.

1. Definitions

Table 1.1. Definitions

Parameter	Abbreviation	Units	Remarks
Calculational system	CS		Infinite grid of multi-assemblies/single assemblies or core
CS symmetry sector	Sim		30 for 30°, 60 for 60°, 120 for 120°, 360 for full CS.
Reactivity of CS	RO	pcm	RO = (Keff-1)/Keff*1.E5
Calculational volume	Vij		Axial fraction j of assembly number i. In VVER-1000 calculations, 10-30 axial fractions of equal volume are usually used.
Effective multiplication factor of CS	Keff		
Multiplication factor of CS	Ko		Relation of neutron generation to neutron absorption. For core calculations Ko values are attributed to Vij
3-D power distribution in core	q_{ij}		Power in Vij normalised by average Vij power
Volume power peaking factor	Kv		Maximum in q_{ij} values
Radial position of volume power peaking factor	N (Kv) or N_K		Number of assembly in calculational core sector where Kv is realised
Axial position of volume power peaking factor	M (Kv) or N_Z		Number of axial level where Kv is realised
3-D burnup distribution in core	BUij	MWd/kg	Burnup in Vij.

		or GWd/t	
2-D power distribution in core	q_i		Assembly powers normalised by average assembly power in core.
Radial power peaking factor	Kq		Maximum in qi values
Radial position of radial power peaking factor	N (Kq) or N_K		Number of assembly in calculational core sector where Kq is realised
Pin linear power	Ql	W/cm	Pin power for 1 cm of an axial calculational fraction
Moment during fuel irradiation	T	EFPD	
2-D burnup distribution in core	BUi	MWd/kg	Average-assembly burnup distribution in core.
Average burnup in Uranium assemblies	$\bar{B}_U$	MWd/kg or GWd/t	
Average burnup in MOX assemblies	$\bar{B}_{MOX}$	MWd/kg or GWd/t	
Average Boron acid (H_3BO_3) concentration[a] in coolant	Cb or C_{H3BO3}	ppm or g/kg	H_3BO_3 fraction in coolant (unit "ppm" means mg of boron acid in 1 Kg of H_2O)
Critical boron acid concentration in coolant	Cb^{crit}	ppm or g/kg	Cb (C_{H3BO3}) value ensuring Keff=1
2-D power distribution in CS	q_k-CS		Power of fuel pins normalised by average fuel pin power in CS.
Peaking factor of 2-D power distribution in CS	K_{FA}–CS		Maximum in q_k-CS values
2-D power distribution in assembly	q_k		Power of fuel pins normalised by average fuel pin power in assembly (in some axial fraction).
3-D power distribution in axial volumes	q_{ijk}		Power of axial volumes of fuel pins normalised

[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.

of fuel pins in core			by average power in such volumes over a whole core
Pin power peaking factor in assembly	Kki		Among q_k values for an assembly number i for a fraction number j where maximum q_{ij} for this assembly is realised.
Radial pin power peaking factor	Kr		max (qi * Kki)
Radial position of radial pin power peaking factor	N (Kr) or N_K		Number of assembly in calculational core sector where Kr is realised
2-D power peaking factor in assembly	K_{FA} (in Russian exploitation calculations the notation Kk or Kk_{max} is also used)		Maximum relative power of fuel pins (maximum in q_k values)
Axial power peaking factor in assembly or in fuel pin	Kz		Maximum relative power of axial volume in assembly or in fuel pin normalised by average power in such volumes (in assembly or in fuel pin)
Total power peaking factor	Ko or $K_{o\text{-}total}$		$\max_{ij} (q_{ij} * Kki) = Kr*Kz$
Radial position of total power peaking factor	N ($K_{o\text{-}total}$) or N_K		Number of assembly in calculational core sector where $K_{o\text{-}total}$ is realised
Axial position of total power peaking factor	M ($K_{o\text{-}total}$) or N_Z		Number of axial level where $K_{o\text{-}total}$ is realised
Engineering factor	K_{eng}		Coefficient taking account of uncertainty of a hot point (maximum fuel pin local power) calculations
2-D burnup distribution in assembly	BUk	MWd/kg or GWd/t	Average-pin burnup distribution in CS.

1-D burnup distribution in fuel pin	BUpin		Burnup distribution in concentric zones of equal volume in fuel pin, normalised by average zone burnup.
1-D power distribution in fuel pin	q_{pin}		Power distribution in concentric zones of equal volume in fuel pin, normalised by average zone power.
Regulation bank position	H_{reg}	cm	Distance from core bottom till rods lower edge
Control rods worth (in core)	$(RO)_{AP-1}$	ppm	Effect of control rods insertion in core supposing the most effective single CR stuck in upper position. It is defined as a reactivity difference in two states: $(RO)_{AP-1}$= RO1-RO2. The second state differs from the first one only by additional CRs inserted in core. All the other parameters correspond to the first state: Cb (that is equal to Cb crit for the first state), temperature and FP distribution in core.
Repeat Criticality Temperature	RCT	°C	Temperature that ensures a secondary critical state during core cooling in EOC in such conditions: all control rods inserted in core except one the most effective, zero boron concentration, equilibrium xenon concentration corresponding to reactor power before its shut-down.
Moderator temperature coefficient (in core)	MTC	pcm/°C	
Moderator density coefficient (in core)	MDC	pcm/g/cc	
Doppler temperature coefficient (in core)	DTC	pcm/°C	Calculated supposing average fuel temperature changing of 1°C
Doppler isotermic temperature coefficient (in core)	DTC*	pcm/°C	Calculated supposing local fuel temperature changing of 1°C

Doppler power coefficient (in core)	DPC	pcm/MW	
Boron reactivity coefficient (in core)	DRO/DCB	pcm/ppm	
Effective fraction of delayed neutrons	βeff or $\beta_{ef.}$	ppm	General characteristic of infinite grid or core
Lifetime of prompt neutrons	λ_m or λ_{im}	s	General characteristic of infinite grid or core
Reactor thermal power	W	MW	
Specific reactor thermal power in CS	Wv	KW/litre	Reactor thermal power in CS volume unit
Nominal reactor thermal power	Wnom	MW	Equal to 3000 MW for VVER-1000
Minimum controllable level of reactor power	MCL	MW	In calculations corresponds to Zero Power and uniform temperature 280°C in core.
Core coolant flow rate	G	m^3/h	
Average entry core temperature	t_{entry}	°C or K	
Average outer core temperature	t_{out}	°C or K	
Average coolant-moderator temperature in CS	t_{mod}	°C or K	
Average Coolant-moderator density in CS	γ_{mod}	g/cm^3	
Fuel temperature	t_{fuel}	K	
Average temperature of other CS components	t_{con}	°C or K	
Fuel pin cladding temperature	t_{clad}	°C or K	
Xenon-135 concentration distribution in core	Xe	10^{24}/cc	For 1 cc in fuel. Xe = 0 → xenon is absent; Xe = 1 → Xe=Xe eq (W).
Equilibrium Xenon-135 concentration distribution in core	Xe eq (W)	10^{24}/cc	Concentration formed during long working with W power, regulating bank in nominal position[b]
Sm-149 concentration distribution in core	Sm	10^{24}/cc	For 1 cc in fuel. Sm = 0 → samarium is absent, Sm = 1 → Sm=Sm eq,

[b] In VVER-1000 calculations Hreg in nominal position is equal to 80% if there is no special indication

			Sm = 3 → full decay of Pm-149 into Sm-149 is simulated in BOC.
Equilibrium Sm-149 concentration distribution in core	Sm eq	10^{24} /cc	Concentration formed during long working, regulating bank in nominal position
Samarium-149 concentration distribution, all Prometium-149 decayed in Sm	Smh	10^{24} /cc	
Core reactivity while reactor shut-down	RO_{STOP}	pcm	Under conditions: W=0, Xe=0,Sm=Smh, t_{mod}= t_{fuel}= t_{con}=20°C, Cb= 16000 ppm

2. Parametric Studies of MOX LTA design (Stage "Assembly")

2.1. Calculational Model. General features

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

- infinite grid of single plutonium or uranium assemblies;
- 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 2.6 (for 100% Plutonium MOX LTA that is not the case of the Report), 2.8 ("Island-1") and 2.10 ("Island-2").

Composition of weapons grade plutonium, adopted for calculations, is presented in Table 2.1. The design parameters of plutonium and uranium assemblies are described in Tables 2.2-2.6.

The calculational model includes two principal regimes described in p.2.1.1 and 2.1.2.

2.1.1 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. A set of calculations for zero irradiation has been executed with Cb=0 and Cb (nat.B)=1200ppm;
- Wv = 108 KW/litre;
- t_{mod} = 302°C;
- t_{con} = 302°C;
- t_{fuel} = 1027 K;
- Xe=Xe eq;
- Sm=Sm eq.

2.1.2. 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;

t_{mod}=t_{con}=t_{fuel}: 20, 280 °C.

2.2. Calculations of «Island» Type MOX LTA. Details

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 2.7);
- two-zones island or "Island-2"(Figure 2.9).

The studies are divided into three parts:

1. Studies of infinite grid of fresh MOX LTA by means of plutonium content variation to ensure an acceptable value of power peaking factor Kk. Axial buckling in this case was variable to provide Keff=1.
2. Calculation of CS where MOX LTA or Uranium FA is surrounded by uranium assemblies, for zoning option chosen in the previous part. In this part plutonium/uranium fuel irradiation has been simulated with fixed axial buckling.
3. 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 (at the stage "Core") could lead to rather low power values q_i in MOX LTA and finally to acceptable values of overcore power peaking factors.

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.

2.2.1. "Island-1" option

The studies for uranium zone enrichment of 3.7% have shown (Figure 2.14) 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 2.15):

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

For the 3% plutonium enrichment Fig.2.24 shows the comparison of inter-assembly row-by-row power distribution for the Uranium zone enrichments of 3.7% and 4.4% with different boron concentrations in coolant Cb (nat) of 0 and 1200 ppm. It is seen that maximum power is attained in Plutonium rods in the last (fifth) "Island" row. The same conclusion can be made from Fig.2.25 with 4% Plutonium central part.

2.2.2. "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 2.14 and 2.15 (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.

The Figures 2.16 and 2.17 show coolant boron concentration influence on optimal values of plutonium enrichment. It is seen that optimal location does not vary significantly.

The Figures 2.18 and 2.19 show row-by-row evolution of maximum relative cell power W. The boron concentration Cb (nat) is equal to 1200 ppm. It is seen from Fig.2.18, that in the case of 4% Plutonium central part, the cell powers in the interior of "Island" exceed the ones in the Uranium region. Besides for the periphery enrichment of 2.5% and 3% the maximum power is located in the forth row and for the periphery enrichment of 3.2%, 3.5% and 4% it is replaced to the fifth row (peripheral "Island" row).

If the Uranium zone enrichment is equal to 4.4% (Figures 2.20 and 2.21) the power in peripheral assembly can exceed the one in the assembly central part as it is seen from the Fig. 2.21 with 3% Plutonium in the "Island" central part. The Figures 2.22 and 2.23 complete this conclusion showing the comparison of different uranium zone MOX LTAs (3.7% and 4.4%). Peripheral enrichment is supposed optimal i.e. with minimum Kk and the central part Plutonium enrichment is of 4% (Fig.2.22) and 3% (Fig.2.23).

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 2.12 and 2.13.

The results of calculations simulating fuel irradiation are presented in Table 2.10 (MOX assembly) and in Tables 2.8 and 2.9 (UOX assembly correspondingly without and with Boron BPRs). Calculations in zero power states are presented in Table 2.7.

2.2.3 "Plutonium island" size variation

Increased size of "Plutonium Island» that comprises 6 plutonium rows (Fig.2.26) has been also considered. In Fig.2.27 and 2.28 the central plutonium enrichment has been fixed by 4% while considering two uranium environment enrichments: 3.7% and 4%. The Figures 2.27 and 2.28 shows an optimum plutonium periphery enrichment about 3% where Kk minimum is reached.

2.2.4 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.2.29-2.40 they are presented for some character pins:

- near central instrumentation tube (as No 77 in Fig.2.18),
- near water tube (as No 76 in Fig.2.18),
- on the border of different «Island-2» enrichments (as No 75 in Fig.2.18),
- on the «Island-2» periphery (as No 74 in Fig.2.18),
- in uranium fuel pin (as No 72 in Fig.2.18).

The following moments while fuel burning have been considered: 0, 12, 24 and 40 MWd/kg.

Figures 2.29 and 2.30 show correspondingly inter-pin relative burnup and power distributions BU_{pin} and q_{pin}. Figures 2.31-2.40 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 that corresponds approximately to fuel discharged after one and three years of reactor exploitation.

2.2.5 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 2.41-2.43 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-2» type MOX LTA design.

Relative power distributions are shown in Figures 2.44 and 2.45 for the following moments while fuel burning 0,12, 24 and 40 MWd/kg.

Relative burnup distributions are shown in Fig.2.46 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.2.47 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.2.48 for a number of plutonium and uranium enrichment compositions.

In Figures 2.49-2.54 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.

3. CALCULATIONS OF VVER-1000 CORE WITH 3 MOX LTAs (Stage "Core")

These studies comprise:

- ***"Uranium Core"***. Calculation of the so-called Advanced VVER-1000 core with boron BPRs for the equilibrium fuel cycle [2] that was defined as basic for 3 MOX LTAs introduction.

- ***"MOX Core"***. Studies of VVER-1000 core with introduction of 3 MOX LTAs of "Island-2" design with the zoning chosen in Chapter 2. Three cycles till MOX LTAs discharge have been studied. Corresponding loading patterns for every cycle have been chosen to minimize power peaking factors.

"Uranium core" loading pattern is shown in Fig.3.1. This figure includes particularly the reloading scheme (the FA locations in previous fuel cycle are indicated), the FA locations in current equilibrium cycle with the indication of its type (according to Figures 2.1, 2.3 and 2.4) and initial average assembly burnups.

The core, FA, fuel pins, CR and Boron BPR geometric and material parameters are indicated in Tables 2.1-2.6.

The reflectors are described in Annex.

3.1. Limitations

Safety limitations

Composed core loading patterns must meet a number of safety requirements.

Tables 3.1 and 3.2 present the requirements that are officially adopted nowadays for VVER-1000 Uranium cores.

For MOX fueled cores the limitations, not yet officially established, have been conventionally strengthened for power peaking factors and RCT. They are presented in Tables 3.3 and 3.4. It was tried to meet these conventional requirements either for MOX LTAs only (it concerns power peaking factors) or for the core (it concerns RCT).

Other limitations

3 MOX LTA are placed in the core under the following conditions:

- respect 120° symmetry;
- not to occupy the positions without in-core measurement system (the self-powered detectors are shown in Fig. 3.6);
- it is desirable to place MOX assemblies symmetrically to the uranium ones that are equipped by detectors.

3.2. Fuel Irradiation Simulation

Irradiation of the fuel loading is simulated with the step 20 EFPD. Cb crit is found in sequence (below these values are named "Cb burnup") until reactivity margin reaches 0, i.e. Cb crit becomes 0. This moment defines T cycle - a value of cycle length usually presented in EFPD unit.

In the process of irradiation:

- Regulating Bank N 10 (Figure 3.6) is 20% inserted in core; other banks are out of core;
- W=Wnom (3000 MW);
- t_{entry} = 287°C;
- Xe=Xe eq;
- At the beginning of irradiation Sm = Smh.

At the stage "MOX core", while studying of acceptable MOX location in the Uranium loading pattern (Fig.3.1), calculations of three successive cycles are carried out with corresponding description of reloading scheme.

3.3. Calculational States

The states that are considered at the stage "Core" are characterized by:

- CRs positions in core (X% N↓ means that the Bank N is X% inserted in core). No indication means that all the CRs are out of the core;
- Cb;
- Average FP concentration in core (Xe-135 and Sm-149 poisoning are considered separately);
- Xe;
- Sm;
- W (in these studies two power levels are considered - W_{nom} и MCL);
- t_{mod};
- t_{fuel};
- t_{con}.

It is necessary to remark that three last parameters are not generally independent.

All the states considered in the process of irradiation will be named "Burn-up".

The specific moments are introduced: the beginning of cycle (BOC) and the end of cycle (EOC). They characterize FP concentration (average in core) in these moments. It should be noted that the other above-mentioned parameters are not always connected directly with irradiation conditions in these moments; their values may depend on reactor start-up conditions before irradiation or cooling conditions in the end of irradiation.

3.4. Information Release

The table below presents the states considered and the parameters calculated. The second column indicates the list of results presented in this report. The rest of calculated parameters and additional information can be received by addressing to Youri Styrine (email: Youri.Styrine@vver.kiae.ru).

Parameter	**Presented in the Report**	**States**							
qi	+		Burn-up						
qij			Burn-up						
qk	+		Burn-up[c]						
Kr	+		Burn-up[c]						
$K_{o\text{-}total}$	+		Burn-up[c]						
Kk i	+		Burn-up[c]						
Ql	+		Burn-up						
BUi	+		Burn-up						
BUij			Burn-up						
BUк			Burn-up						
MTC	+		Burn-up	BOC, MCL, Xe=0, t_{mod}= t_{fuel}= t_{con}= 280°C, Cb crit	EOC, MCL, Xe=Xe eq, t_{mod}= t_{fuel}= t_{con}= 280°C, Cb crit				
MDC	+		Burn-up	BOC, MCL, Xe=0, t_{mod}= t_{fuel}= t_{con}= 280°C, Cb crit	EOC, MCL, Xe=Xe eq, t_{mod}= t_{fuel}= t_{con}= 280°C, Cb crit				

[c] For MOX assemblies and for an assembly with maximum qi.
[c] For MOX assemblies and for an assembly with maximum qi.
[c] For MOX assemblies and for an assembly with maximum qi.
[c] For MOX assemblies and for an assembly with maximum qi.

DTC	+		Burn-up	BOC, MCL, Xe=0, t_{mod}= t_{fuel}= t_{con}= 280°C, Cb crit	EOC, MCL, Xe=Xe eq, t_{mod}= t_{fuel}= t_{con}= 280°C, Cb crit				
DRO/DCB	+		Burn-up	BOC, MCL, Xe=0, t_{mod}= t_{fuel}= t_{con}= 280°C, Cb crit	EOC, MCL, Xe=Xe eq, t_{mod}= t_{fuel}= t_{con}= 280°C, Cb crit				
βeff and λm	+		Burn-up	BOC, MCL, Xe=0, t_{mod}= t_{fuel}= t_{con}= 280°C, Cb crit	EOC, MCL, Xe=Xe eq, t_{mod}= t_{fuel}= t_{con}= 280°C, Cb crit				
Cb crit	+		Burn-up	BOC, MCL, Xe=0, t_{mod}= t_{fuel}= t_{con}= 280°C, Cb crit	EOC, MCL, Xe=Xe eq, t_{mod}= t_{fuel}= t_{con}= 280°C, Cb crit				
RO stop	+	W=0, Xe=0, Sm=Smh t_{mod}= t_{fuel}= t_{con}= 20°C , Cb = 16000 ppm							

RCT	+	EOC, MCL, Xe=Xe eq, t_{mod}= t_{fuel}= t_{con}= 280°C, Cb = 0, 100% 1-10↓ (except of the most effective single CR)							
$(RO)_{AP-1}$	+	**S1**:BOC, Wnom, Xe=Xe eq, t_{entry}=287°C, Cb burnup 100 % 5↓ 30 % 10↓ **S2**[b]: the same but 100% 1-10↓	**S1**:BOC, MCL, Xe=0, t_{entry}=280°C Cb crit 30% 10↓ **S2**: the same but 100% 1-10↓	**S1**:BOC, MCL, Xe=Xe eq, t_{entry}=280°C Cb crit 30% 10↓ **S2**: the same but 100% 1-10↓	**S1**:EOC, Wnom, Xe=Xe eq, t_{entry}=287°C Cb burnup 100 % 5↓ 30% 10↓ **S2**: the same but 100% 1-10↓	**S1**:EOC, MCL, Xe=Xe eq, t_{entry}=280°C Cb crit 100 % 5↓ 30 % 10↓ **S2**: the same but 100% 1-10↓	**S1**:EOC, MCL, Xe=0, t_{entry}=280°C Cb crit 100 % 5↓ 30 % 10↓ **S2**: the same but 100% 1-10↓	**S1**:BOC, Wnom, Xe=Xe eq, t_{entry}=287°C, Cb burnup 20 % 10↓ **S2**: the same but with successive introduction of the Banks 1-9 (0%↓, 10%↓, 20%↓... 100%↓)	**S1**:EOC, Wnom, Xe=Xe eq, t_{entry}=287°C Cb burnup 20 % 10↓ **S2**: the same but with successive introduction of the Banks 1-9 (0%↓, 10%↓, 20%↓... 100%↓)

[b] For all the states S2 : the most effective single CR is supposed stuck in upper position.

3.5. Calculational Results

3.5.1 Uranium Core

The Table 3.5 and Fig. 3.1 show the results of kinetics parameters calculations for the equilibrium fuel cycle in the Uranium base core that have been performed by the code BIPR-7A[a].

The attained power peaking factors obtained by pin-by-pin code PERMAK-A are presented in Table 3.13. The linear pin powers for BOC and EOC are presented correspondingly in Figures 3.2 and 3.3. It is seen from combination of BIPR-7A and PERMAK-A calculations that maximum linear pin power in BOC is attained on level 4[b], in EOC – on level 2. It justifies PERMAK-A calculations to be performed as usual on level 4 (more details about PERMAK-A calculational scheme are described in Annex).

Pin-by-pin power distributions in the most powered assemblie for BOC and EOC are presented correspondingly in Figures 3.4 and 3.5.

Table 3.6 shows the parameters values in zero power states calculated by the code BIPR-7A.

It is seen that Uranium core meets the safety requirements presented in Tables 3.1 and 3.2 for power peaking factors and reactivity coefficients.

Table 3.15a and 3.15b show the CRs worth calculated with certain conservatism (the lowest possible position of Bank 5 that serves for offset regulation and of regulating Bank 10). It is seen that the limiting value of 5500 pcm is respected.

Table 3.16 shows core reactivity evolution in the process of control rods simultaneous movement (when AP is actuated) from top to the bottom of core. BOC and EOC moments are considered including the situations when the most effective single control rod is stuck in upper position. In initial position all the banks except of Regulating bank 10 were in the upper position.

Table 3.17 shows the RCT value that is essentially lower than the allowable one in Table 3.1.

Table 3.14 describes the scheme of conservative evaluation of core subcriticality (scram margin) after scram actuation and reactor state transformation from nominal power to MCL. The effects and uncertainties involved in this scheme (vapor effect, absorbent irradiation, uncertainty of CRs worth calculation etc.) correspond to ones adopted in the West, particularly, in the US and France.

---

[a] Temperature drop in Fig.3.1 is the difference between output and input coolant temperatures for an assembly considered as a channel.

[b] It should be reminded that the level numeration begins from the core bottom and the number of calculational levels in BIPR-7A was 10.

3.5.2. MOX Core

3 MOX assemblies have been located in uranium reference core according to the principals mentioned in p.3.1.

The positions 8, 88 and 150 for the first MOX loading (Fig.3.7) have been chosen because they possess self-powered detectors (see Fig.3.6). Other assemblies have been replaced to ensure a minimum value of Kq calculated by BIPR-7A. Besides, several fresh assemblies of "Ba" type (it is described in Fig.2.3) have been added to the first MOX loading. Reloading schemes for second and third cycles with 3 MOX LTAs of "Island-2" type are presented correspondingly in Figures 3.17 and 3.27.

The values of average assembly parameters calculated by the code BIPR-7A are presented for 3 successive fuel cycles in Figures 3.8-3.10 and Tables 3.7 (first cycle), Figures 3.18-3.20 and Tables 3.9 (second cycle), Figures 3.28-3.30 and Tables 3.11 (third cycle).

The attained power peaking factors obtained by pin-by-pin code PERMAK-A are presented in Table 3.13. The linear pin powers for BOC and EOC are presented correspondingly in Figures 3.11 and 3.12 (first cycle), Figures 3.21 and 3.22 (second cycle), Figures 3.31 and 3.32 (third cycle). Pin-by-pin power distributions in BOC and EOC both for the most powered assemblies and for MOX LTAs are presented in Figures 3.13-3.16 (first cycle), 3.23-3.26 (second cycle), 3.33-3.36 (third cycle).

Table 3.8, 3.10 and 3.12 show correspondingly the parameters values in zero power states for the first, the second and the third fuel MOX cycles calculated by the code BIPR-7A.

It is seen that MOX cores meet the safety requirements presented in Tables 3.1-3.4 for power peaking factors and reactivity coefficients.

Table 3.15a and 3.15b show the CRs worth. It is seen that the conventional limiting value of 5500 pcm (Table 3.3) is respected.

Table 3.16 shows core reactivity evolution in the process of AP actuation.

Table 3.17 shows the RCT values that are strongly lower than the conventional allowable value of 210°C.

Table 3.14 describes the scheme of conservative evaluation of core subcriticality (scram margin).

It can be seen that the presence of 3 MOX LTAs does not influence $(RO)_{AP}$ in clear manner. Its value is determined first of all by core loading pattern. It may be supposed that only significant value of MOX assemblies in core could lead to lowering of control rods worth because of strong absorbing capacity of MOX fuel.

CONCLUSION

The report presents the results of design studies of "Island" type MOX LTA:

- Parametric studies to define MOX LTA structure primarily to choose plutonium content in assembly zones that ensures reasonable power peaking factors and power generation equivalence in MOX and UOX assemblies.
- Studies of VVER-1000 core characteristics with 3 MOX LTAs introduced for three successive fuel cycles.

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

- uniform «Island» or "Island-1" option;
- graded «Island» with lower plutonium content in one peripheral row of pins or "Island-2" option.

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 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.

VVER-1000 core with boron burnable control rods has been chosen as a base for 3 MOX LTAs introduction.

Fuel loadings with 3 MOX LTAs have been optimized to ensure a minimum value of power peaking factor Kq.

Evolution of main neutronics parameters during 3 successive cycles with MOX LTAs is presented. It is shown that MOX loaded cores meet the safety requirements preliminary adopted for MOX fuel concerning power peaking factors, reactivity coefficients and control rods worth.

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Table 2.1. 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 2.2. 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	%	80
Core Coolant Flow Rate	m^3/hr	84000
Pressure at Core Inlet	MPa	15.7
Core Inlet Temperature	°C	287

Table 2.3. 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 2.4. 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 2.5. 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)
Mass of MOX fuel in Fuel Pin	kg	1.600

Compositions Weight percent:

*

Zr	Nb	Hf
98.97	1.0	0.03

Table 2.6. 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	
		0.036	0.065
B10	Wt%	0.2279	0.4046
B11		1.0153	1.8028
Al		91.7424	88.5951
Fe		0.1915	0.1850
Ni		1.9153	1.8496
Cr		2.9923	5.3133
Zr		1.9153	1.8496

Compositions Weight percent:

*

Zr	Nb	Hf
98.97	1.0	0.03

Table 2.7. 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-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 2.8. 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	1.0370	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 2.9. Parameters Evolution in the Process of Fuel Irradiation. Reference Uranium Assemblage with Boron BPRs

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 2.10. Parameters Evolution in the Process of Fuel Irradiation. "Island-2" Type MOX LTA

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

Table 3.1. Limiting parameters for VVER-1000

Criterion	Limiting Value	Remarks
Kq	<=1.35	For nominal power W=3000 MW
Kr	<=1.60	For nominal power W=3000 MW
$K_{o\text{-}total}$	Tabl. 3.2	For nominal power W=3000 MW
MTC	< 0	
MDC	> 0	
RO stop	<= -2000 pcm	t=20°C, Xe=0, Sm=Smh, Cb=16000 ppm, all control rods extracted
RCT	< 220°C	
$(RO)_{AP\text{-}1}$	> 5500 pcm	In full power

Table 3.2. Limits recommended for total power peaking factor $K_{o\text{-}total}$ for VVER-1000

Layer (from bottom to top)	1	2	3	4	5	6	7	8	9	10
$K_{o\text{-}total}$	2.24	2.24	2.24	2.24	2.24	2.14	1.96	1.80	1.69	1.58

Table 3.3. Recommended limiting parameters for VVER-1000 with 3 MOX LTAs.

Criterion	Limiting Value	Remarks
Kq	<=1.35	
Kr	<=1.55	In MOX assemblies. For nominal power W=3000 MW
$K_{o\text{-}total}$	Tabl. 3.4	In MOX assemblies. For nominal power W=3000 MW
MTC	< 0	
MDC	> 0	
RO stop	<= -2000 pcm	t=20°C, Xe=0, Sm=Smh, Cb=16000 ppm, all control rods extracted
RCT	< 210°C	
$(RO)_{AP\text{-}1}$	> 5500 pcm	In full power

Table 3.4. Limits recommended for total power peaking factor $K_{o\text{-}total}$ in MOX assemblies for VVER-1000 with 3 MOX LTAs

Layer (from bottom to top)	1	2	3	4	5	6	7	8	9	10
$K_{o\text{-}total}$	2.17	2.17	2.17	2.17	2.17	2.07	1.90	1.74	1.64	1.53

Table 3.5. Evolution of main neutronics parameters in Uranium reference core . Equilibrium cycle

Sim = 60 , Xe = 1 , Sm = 3

№	T EFPD	$H_{reg.}$ cm	t_{entry} °C	W MW	$Cb^{crit.}$ ppm	G m^3/h	Kq	Nk	Kq^{MOX}	Nk	Kv	Nk	Nz	$\bar{B}_U$ MW• d/kg	$\bar{B}_{MOX}$ MW• d/kg	MDC pcm• $(g/cm^3)^{-1}$	MTC pcm• $°C^{-1}$	DTC pcm• $°C^{-1}$	DTC* pcm• $°C^{-1}$	DPC pcm• MW^{-1}	DRo/DCb pcm• ppm^{-1}	$\beta_{ef.}$ pcm	l_{im} •10^5 sec
1	0.0	283.2	287.0	3000	5657	84000	1.31	19	0.00	0	1.61	19	4	14.14	0.00	12293	−25.94	−2.96	−2.46	−0.29	−1.55	650	2.24
2	20.0	283.2	287.0	3000	5318	84000	1.31	19	0.00	0	1.58	19	4	15.00	0.00	12894	−26.94	−2.96	−2.47	−0.29	−1.55	639	2.24
3	40.0	283.2	287.0	3000	4899	84000	1.31	19	0.00	0	1.56	19	4	15.85	0.00	14000	−29.20	−2.94	−2.48	−0.29	−1.56	630	2.25
4	60.0	283.2	287.0	3000	4473	84000	1.31	19	0.00	0	1.53	19	3	16.70	0.00	15191	−31.69	−2.93	−2.50	−0.29	−1.57	622	2.27
5	80.0	283.2	287.0	3000	4047	84000	1.31	19	0.00	0	1.52	19	3	17.55	0.00	16400	−34.24	−2.93	−2.52	−0.29	−1.58	613	2.29
6	100.0	283.2	287.0	3000	3631	84000	1.31	19	0.00	0	1.51	19	3	18.41	0.00	17590	−36.77	−2.94	−2.55	−0.29	−1.59	606	2.31
7	120.0	283.2	287.0	3000	3215	84000	1.30	19	0.00	0	1.50	19	3	19.26	0.00	18775	−39.30	−2.96	−2.58	−0.29	−1.60	598	2.33
8	140.0	283.2	287.0	3000	2813	84000	1.30	19	0.00	0	1.49	19	3	20.11	0.00	19928	−41.77	−2.97	−2.60	−0.29	−1.62	591	2.35
9	160.0	283.2	287.0	3000	2411	84000	1.30	19	0.00	0	1.48	19	3	20.96	0.00	21077	−44.25	−2.99	−2.63	−0.29	−1.63	585	2.37
10	180.0	283.2	287.0	3000	2023	84000	1.30	19	0.00	0	1.47	19	2	21.82	0.00	22203	−46.69	−3.02	−2.66	−0.29	−1.64	578	2.40
11	200.0	283.2	287.0	3000	1634	84000	1.30	19	0.00	0	1.47	19	2	22.67	0.00	23333	−49.16	−3.04	−2.69	−0.29	−1.66	573	2.42
12	220.0	283.2	287.0	3000	1254	84000	1.29	19	0.00	0	1.47	19	2	23.52	0.00	24457	−51.62	−3.06	−2.71	−0.29	−1.67	567	2.45
13	240.0	283.2	287.0	3000	874	84000	1.29	19	0.00	0	1.47	19	2	24.37	0.00	25592	−54.13	−3.08	−2.74	−0.30	−1.68	562	2.48
14	260.0	283.2	287.0	3000	500	84000	1.29	19	0.00	0	1.46	19	2	25.23	0.00	26727	−56.64	−3.09	−2.76	−0.30	−1.70	557	2.51
15	280.0	283.2	287.0	3000	127	84000	1.28	19	0.00	0	1.46	19	2	26.08	0.00	27869	−59.18	−3.11	−2.79	−0.30	−1.71	552	2.54
16	286.9	283.2	287.0	3000	0	84000	1.28	19	0.00	0	1.45	19	2	26.37	0.00	28260	−60.05	−3.12	−2.80	−0.30	−1.72	551	2.55

Table 3.6. Main neutronics parameters in zero power states. Reference Uranium Core Equilibrium Cycle

T	RO pcm	Cb ppm	Bank 10	Other banks↓↑	Xe	Sm	Tmod °C	MTC pcm/°C	MDC pcm/g/cc	DTC pcm/°C	DRO/DCB pcm/ppm	λm $*10^5$s	βeff *100
BOC	0	8860	100% ↑	100% ↑	0	Smh	280	-1.23	2210	-2.93	-1.49	2.10	0.65
EOC	0	2000	100% ↑	100% ↑	eq	Sm eq	280	-27.52	18730	-3.31	-1.76	2.44	0.57
BOC	-14237 (RO_{STOP})	16000	100% ↑	100% ↑	0	Smh	20						

Table 3.7. Evolution of main neutronics parameters. First cycle with 3 MOX LTAs of "Island-2" type

Sim =360 , Xe = 1 , Sm = 3

№	T EFPD	$H_{reg.}$ cm	t_{entry} °C	W MW	$Cb^{crit.}$ ppm	G m^3/h	Kq	Nk	Kq^{MOX}	Nk	Kv	Nk	Nz	$\bar{B}_U$ MW• d/kg	$\bar{B}_{MOX}$ MW• d/kg	MDC pcm• $(g/cm^3)^{-1}$	MTC pcm• $°C^{-1}$	DTC pcm• $°C^{-1}$	DTC* pcm• $°C^{-1}$	DPC pcm• MW^{-1}	DRo/DCb pcm• ppm^{-1}	$\beta_{ef.}$ pcm	l_{im} •10^5 sec
1	0.0	283.2	287.0	3000	5773	84000	1.32	38	1.01	8	1.61	38	4	14.26	0.00	11944	−24.84	−2.88	−2.49	−0.28	−1.57	647	2.25
2	20.0	283.2	287.0	3000	5435	84000	1.27	38	0.97	8	1.52	38	4	15.12	0.86	12535	−25.79	−2.88	−2.50	−0.28	−1.57	636	2.25
3	40.0	283.2	287.0	3000	5014	84000	1.26	38	0.97	8	1.49	38	4	15.97	1.69	13669	−28.14	−2.87	−2.51	−0.28	−1.57	628	2.27
4	60.0	283.2	287.0	3000	4586	84000	1.26	117	0.97	8	1.47	47	3	16.82	2.52	14879	−30.69	−2.87	−2.53	−0.28	−1.59	620	2.28
5	80.0	283.2	287.0	3000	4158	84000	1.26	72	0.96	150	1.45	72	3	17.67	3.34	16104	−33.29	−2.88	−2.55	−0.28	−1.60	612	2.30
6	100.0	283.2	287.0	3000	3737	84000	1.26	72	0.96	150	1.44	132	3	18.53	4.16	17315	−35.88	−2.89	−2.58	−0.28	−1.61	604	2.32
7	120.0	283.2	287.0	3000	3316	84000	1.26	132	0.96	88	1.44	132	3	19.38	4.98	18523	−38.47	−2.90	−2.60	−0.28	−1.62	597	2.34
8	140.0	283.2	287.0	3000	2905	84000	1.26	132	0.96	88	1.44	132	3	20.23	5.80	19708	−41.02	−2.92	−2.62	−0.28	−1.63	590	2.36
9	160.0	283.2	287.0	3000	2493	84000	1.26	132	0.96	88	1.43	124	3	21.09	6.62	20889	−43.58	−2.94	−2.65	−0.28	−1.64	584	2.39
10	180.0	283.2	287.0	3000	2093	84000	1.27	132	0.96	88	1.44	124	2	21.94	7.44	22050	−46.11	−2.96	−2.67	−0.29	−1.66	578	2.41
11	200.0	283.2	287.0	3000	1694	84000	1.27	124	0.96	88	1.44	124	2	22.79	8.25	23214	−48.66	−2.98	−2.70	−0.29	−1.67	572	2.44
12	220.0	283.2	287.0	3000	1301	84000	1.27	124	0.96	88	1.45	124	2	23.65	9.07	24372	−51.19	−3.00	−2.72	−0.29	−1.68	566	2.47
13	240.0	283.2	287.0	3000	909	84000	1.27	124	0.96	88	1.45	124	2	24.50	9.88	25537	−53.76	−3.02	−2.74	−0.29	−1.70	561	2.49
14	260.0	283.2	287.0	3000	524	84000	1.27	124	0.96	88	1.45	124	2	25.35	10.70	26697	−56.33	−3.04	−2.76	−0.29	−1.71	556	2.52
15	280.0	283.2	287.0	3000	139	84000	1.27	124	0.96	88	1.45	124	2	26.21	11.51	27861	−58.91	−3.05	−2.79	−0.29	−1.73	552	2.55
16	287.4	283.2	287.0	3000	0	84000	1.27	124	0.96	88	1.45	124	2	26.52	11.81	28287	−59.87	−3.06	−2.79	−0.29	−1.73	550	2.57

Table 3.8. Main neutronics parameters in zero power states. First cycle with 3 MOX LTAs of "Island-2" type

T	RO pcm	Cb ppm	Bank 10	Other banks↓↑	Xe	Sm	Tmod °C	MTC pcm/°C	MDC pcm/g/cc	DTC pcm/°C	DRO/DCB pcm/ppm	λm $*10^5$s	βeff *100
BOC	0	88900	100% ↑	100% ↑	0	Smh	280	-0.75	2090	-2.96	-1.50	2.11	0.65
EOC	0	1960	100% ↑	100% ↑	eq	Sm eq	280	-27.64	18840	-3.31	-1.78	2.46	0.56
BOC	-14338 (RO_{STOP})	16000	100% ↑	100% ↑	0	Smh	20						

Table 3.9. Evolution of main neutronics parameters. Second cycle with 3 MOX LTAs of "Island-2" type

Sim =360 , Xe = 1 , Sm = 3

№	T EFPD	$H_{reg.}$ cm	t_{entry} °C	W MW	$Cb^{crit.}$ ppm	G m^3/h	Kq	Nk	Kq^{MOX}	Nk	Kv	Nk	Nz	$\bar{B}_U$ MW• d/kg	$\bar{B}_{MOX}$ MW• d/kg	MDC pcm• $(g/cm^3)^{-1}$	MTC pcm• $°C^{-1}$	DTC pcm• $°C^{-1}$	DTC* pcm• $°C^{-1}$	DPC pcm• MW^{-1}	DRo/DCb pcm• ppm^{-1}	$\beta_{ef.}$ pcm	l_{im} $•10^5$ sec
1	0.0	283.2	287.0	3000	5658	84000	1.34	153	1.23	141	1.66	153	4	13.86	11.81	12366	−25.86	−2.87	−2.47	−0.28	−1.57	647	2.25
2	20.0	283.2	287.0	3000	5322	84000	1.28	153	1.23	141	1.55	153	4	14.70	12.86	12989	−26.89	−2.88	−2.49	−0.28	−1.57	636	2.25
3	40.0	283.2	287.0	3000	4905	84000	1.28	153	1.22	141	1.52	153	4	15.55	13.91	14105	−29.20	−2.87	−2.51	−0.28	−1.57	628	2.27
4	60.0	283.2	287.0	3000	4487	84000	1.27	153	1.21	141	1.49	153	3	16.40	14.95	15283	−31.67	−2.87	−2.53	−0.28	−1.58	619	2.28
5	80.0	283.2	287.0	3000	4061	84000	1.27	153	1.20	141	1.47	153	3	17.25	15.98	16492	−34.24	−2.87	−2.55	−0.28	−1.59	612	2.30
6	100.0	283.2	287.0	3000	3641	84000	1.26	153	1.20	18	1.45	47	3	18.10	17.00	17687	−36.78	−2.88	−2.57	−0.28	−1.61	604	2.32
7	120.0	283.2	287.0	3000	3221	84000	1.25	153	1.19	18	1.43	47	3	18.95	18.03	18878	−39.34	−2.90	−2.60	−0.28	−1.62	597	2.34
8	140.0	283.2	287.0	3000	2817	84000	1.24	47	1.19	18	1.41	47	3	19.80	19.04	20037	−41.83	−2.91	−2.62	−0.28	−1.63	590	2.36
9	160.0	283.2	287.0	3000	2413	84000	1.24	110	1.18	18	1.40	110	3	20.65	20.05	21192	−44.32	−2.93	−2.65	−0.28	−1.64	584	2.38
10	180.0	283.2	287.0	3000	2016	84000	1.24	110	1.18	18	1.40	110	2	21.50	21.06	22334	−46.80	−2.95	−2.67	−0.29	−1.65	578	2.41
11	200.0	283.2	287.0	3000	1620	84000	1.25	110	1.17	18	1.41	110	2	22.35	22.07	23479	−49.29	−2.97	−2.70	−0.29	−1.67	572	2.44
12	220.0	283.2	287.0	3000	1234	84000	1.25	110	1.17	18	1.42	110	2	23.20	23.06	24610	−51.76	−2.99	−2.72	−0.29	−1.68	566	2.46
13	240.0	283.2	287.0	3000	849	84000	1.25	110	1.17	18	1.42	110	2	24.05	24.06	25749	−54.26	−3.01	−2.74	−0.29	−1.69	561	2.49
14	260.0	283.2	287.0	3000	469	84000	1.25	110	1.16	18	1.42	110	2	24.90	25.05	26885	−56.76	−3.03	−2.76	−0.29	−1.71	556	2.52
15	280.0	283.2	287.0	3000	90	84000	1.25	110	1.16	18	1.42	110	2	25.75	26.05	28028	−59.29	−3.04	−2.79	−0.29	−1.72	552	2.55
16	284.8	283.2	287.0	3000	0	84000	1.25	110	1.16	18	1.42	56	2	25.95	26.28	28301	−59.90	−3.05	−2.79	−0.29	−1.73	551	2.56

Table 3.10. Main neutronics parameters in zero power states. Second cycle with 3 MOX LTAs of "Island-2" type

T	RO pcm	Cb ppm	Bank 10	Other banks↓↑	Xe	Sm	Tmod °C	MTC pcm/°C	MDC pcm/g/cc	DTC pcm/°C	DRO/DCB pcm/ppm	λm *10^5 s	βeff *100
BOC	0	9330	100% ↑	100% ↑	0	Smh	280	-1.61	2540	-2.96	-1.51	2.12	0.65
EOC	0	2090	100% ↑	100% ↑	eq	Sm eq	280	-27.85	18940	-3.31	-1.77	2.45	0.56
BOC	-14463 (RO_{STOP})	16000	100% ↑	100% ↑	0	Smh	20						

Table 3.11. Evolution of main neutronics parameters. 3-d cycle with 3 MOX LTAs of "Island-2" type

Sim =360 , Xe = 1 , Sm = 3

№	T EFPD	$H_{reg.}$ cm	t_{entry} °C	W MW	$Cb^{crit.}$ ppm	G m^3/h	Kq	Nk	Kq^{MOX}	Nk	Kv	Nk	Nz	$\bar{B}_U$ MW•d/kg	$\bar{B}_{MOX}$ MW•d/kg	MDC pcm•$(g/cm^3)^{-1}$	MTC pcm•°C^{-1}	DTC pcm•°C^{-1}	DTC* pcm•°C^{-1}	DPC pcm•MW^{-1}	DRo/DCb pcm•ppm^{-1}	$\beta_{ef.}$ pcm	l_{im} •10^5 sec
1	0.0	283.2	287.0	3000	5790	84000	1.33	126	1.03	111	1.64	126	4	13.41	26.28	11833	−24.63	−2.89	−2.49	−0.28	−1.56	648	2.24
2	20.0	283.2	287.0	3000	5455	84000	1.28	126	1.06	111	1.54	126	4	14.26	27.16	12483	−25.71	−2.89	−2.50	−0.28	−1.56	638	2.25
3	40.0	283.2	287.0	3000	5039	84000	1.27	11	1.05	111	1.51	126	4	15.11	28.06	13606	−28.04	−2.89	−2.52	−0.28	−1.57	629	2.26
4	60.0	283.2	287.0	3000	4616	84000	1.27	124	1.05	111	1.48	124	4	15.97	28.96	14802	−30.56	−2.88	−2.54	−0.28	−1.58	621	2.28
5	80.0	283.2	287.0	3000	4193	84000	1.27	124	1.05	111	1.47	124	3	16.82	29.85	16012	−33.13	−2.89	−2.56	−0.28	−1.59	613	2.29
6	100.0	283.2	287.0	3000	3770	84000	1.27	124	1.04	111	1.46	124	3	17.67	30.74	17220	−35.70	−2.89	−2.58	−0.28	−1.60	606	2.31
7	120.0	283.2	287.0	3000	3361	84000	1.27	124	1.04	111	1.45	124	3	18.52	31.63	18399	−38.23	−2.90	−2.60	−0.28	−1.61	599	2.33
8	140.0	283.2	287.0	3000	2952	84000	1.27	124	1.04	111	1.44	124	3	19.37	32.52	19573	−40.75	−2.92	−2.63	−0.28	−1.63	592	2.36
9	160.0	283.2	287.0	3000	2543	84000	1.26	124	1.05	111	1.44	124	3	20.23	33.41	20743	−43.28	−2.94	−2.65	−0.28	−1.64	585	2.38
10	180.0	283.2	287.0	3000	2147	84000	1.26	124	1.05	111	1.43	124	2	21.08	34.30	21889	−45.77	−2.95	−2.67	−0.29	−1.65	579	2.40
11	200.0	283.2	287.0	3000	1752	84000	1.26	124	1.05	111	1.44	124	2	21.93	35.19	23039	−48.27	−2.97	−2.69	−0.29	−1.66	573	2.43
12	220.0	283.2	287.0	3000	1357	84000	1.26	124	1.05	111	1.44	124	2	22.78	36.09	24194	−50.80	−2.99	−2.72	−0.29	−1.68	568	2.46
13	240.0	283.2	287.0	3000	974	84000	1.26	124	1.05	111	1.44	124	2	23.63	36.98	25334	−53.30	−3.01	−2.74	−0.29	−1.69	563	2.49
14	260.0	283.2	287.0	3000	592	84000	1.26	124	1.05	111	1.44	124	2	24.48	37.88	26482	−55.84	−3.03	−2.76	−0.29	−1.71	558	2.51
15	280.0	283.2	287.0	3000	210	84000	1.26	124	1.06	111	1.44	124	2	25.34	38.78	27637	−58.39	−3.04	−2.78	−0.29	−1.72	553	2.54
16	291.2	283.2	287.0	3000	0	84000	1.26	124	1.06	111	1.43	124	2	25.81	39.28	28277	−59.82	−3.05	−2.79	−0.29	−1.73	551	2.56

Table 3.12. Main neutronics parameters in zero power states. Third cycle with 3 MOX LTAs of "Island-2" type

T	RO pcm	Cb ppm	Bank 10	Other banks↓↑	Xe	Sm	Tmod °C	MTC pcm/°C	MDC pcm/g/cc	DTC pcm/°C	DRO/DCB pcm/ppm	λm $*10^5$s	βeff *100
BOC	0	8890	100% ↑	100% ↑	0	Smh	280	-0.84	2090	-2.96	-1.50	2.11	0.65
EOC	0	1930	100% ↑	100% ↑	eq	Sm eq	280	-27.84	18940	-3.31	-1.78	2.45	0.56
BOC	-14285 (RO_{STOP})	16000	100% ↑	100% ↑	0	Smh	20						

Table 3.13. Pin Power Peaking Factors Attained During Fuel Cycle

T, EFPD	Kr							N (Kr)				Ko-total				N (Ko-total)				M(Ko-total)			
	UOX	MOX 1		MOX 2		MOX 3		UOX	MOX 1	MOX 2	MOX 3	UOX	MOX 1	MOX 2	MOX 3	UOX	MOX 1	MOX 2	MOX 3	UOX	MOX 1	MOX 2	MOX 3
	ALL CORE	ALL CORE	MOX FA	ALL CORE	MOX FA	ALL CORE	MOX FA					ALL CORE	ALL CORE	ALL CORE	ALL CORE								
0	1.51	1.47	1.40	1.52	1.52	1.48	1.27	19	38	141	126	1.86	1.79	1.84	1.82	19	38	153	126	4	4	4	4
20	1.49	1.40	1.33	1.50	1.50	1.42	1.29	19	38	141	124	1.80	1.68	1.77	1.71	19	38	141	124	4	4	4	4
40	1.48	1.40	1.30	1.48	1.48	1.41	1.28	19	40	141	124	1.76	1.65	1.72	1.67	19	38	141	124	4	4	4	4
60	1.47	1.39	1.29	1.46	1.46	1.40	1.27	19	40	141	124	1.72	1.62	1.66	1.63	19	47	18	124	3	3	4	4
80	1.45	1.38	1.27	1.44	1.44	1.39	1.26	19	72	18	124	1.69	1.59	1.63	1.60	19	132	18	124	3	3	3	3
100	1.44	1.37	1.26	1.42	1.42	1.37	1.26	19	72	18	124	1.66	1.58	1.60	1.58	19	72	18	124	3	3	3	3
120	1.43	1.37	1.25	1.41	1.41	1.36	1.24	19	72	18	124	1.64	1.57	1.57	1.56	19	72	18	124	3	3	3	3
140	1.42	1.36	1.24	1.40	1.40	1.35	1.24	19	72	18	124	1.62	1.55	1.55	1.54	19	124	18	124	3	3	3	3
160	1.41	1.36	1.23	1.38	1.38	1.34	1.23	19	72	18	124	1.60	1.54	1.53	1.52	19	124	18	124	3	3	2	3
180	1.39	1.35	1.22	1.37	1.37	1.33	1.23	19	124	18	124	1.58	1.54	1.52	1.51	19	124	18	124	3	2	2	2
200	1.38	1.35	1.21	1.36	1.36	1.32	1.23	19	124	18	124	1.57	1.53	1.51	1.50	19	124	18	124	2	2	2	2
220	1.37	1.34	1.20	1.35	1.35	1.32	1.22	19	124	18	124	1.56	1.53	1.50	1.50	19	124	110	124	2	2	2	2
240	1.36	1.34	1.20	1.33	1.33	1.31	1.22	19	124	18	124	1.55	1.52	1.49	1.49	19	124	110	124	2	2	2	2
260	1.35	1.33	1.19	1.32	1.32	1.30	1.22	19	124	18	124	1.54	1.52	1.49	1.48	19	124	110	124	2	2	2	2
280	1.34	1.32	1.19	1.31	1.31	1.30	1.21	6	124	18	124	1.53	1.51	1.48	1.48	19	124	56	124	2	2	2	2
EOC	1.34	1.32	1.19	1.31	1.31	1.30	1.21	6	124	18	124	1.52	1.51	1.48	1.47	19	124	56	124	2	2	2	2

(shaded cell) Power peaking factor is attained in MOX LTA

Table 3.14. Core Subcriticality (Scram Margin) in different states in the process of Scram actuation

State parameters						RO, pcm							
State Number	W, MW	t_{entry}, °C	Hreg, %	Position of banks 1-9, %	Position of the most eff. CR, %	UOX		MOX 1er cycle		MOX 2nd cycle		MOX 3d cycle	
						BOC	EOC	BOC	EOC	BOC	EOC	BOC	EOC
1	3000	Nominal.	100	100	100	+522	+605	+484	+597	+432	+561	+453	+575
	Regulation margin of reactivity												
2	3000	Nominal.	50	100	100	0.	0.	0.	0.	0.	0.	0.	0.
	Scram actuation without sticking of the most effective CR												
3	3000	Nominal.	0	0	0	-8833	-9136	-8782	-9043	-8819	-9076	-9009	-9151
	Scram actuation with sticking of the most effective CR												
4	3000	Nominal.	0	0	100	-7970	-8262	-7965	-8181	-7900	-8164	-8681	-8271
	Doppler effect												
5	0	Nominal.	0	0	100	-6391	-6807	-6990	-7303	-6879	-7256	-7640	-7376
	Moderator temperature effect												
6	0	287	0	0	100	-5550	-5088	-5718	-5023	-5636	-5027	-6528	-5196
	Moderator temperature effect												
7	0	280	0	0	100	-5358	-4711	-5530	-4647	-5445	-4652	-6343	-4827
	Vapor effect ($\Delta\rho$ = 50 pcm)												
8	0	280	0	0	100	-5308	-4661	-5480	-4597	-5395	-4602	-6293	-4777
	Uncertainty of $(RO)_{AP}$ calculation (10% of p. 4)												
9	0	280	0	0	100	-4511	-3835	-4684	-3779	-4605	-3786	-5425	-3950
	Uncertainty of temperature effect calculation ($\Delta\rho$ = 180 pcm)												
10	0	280	0	0	100	-4331	-3655	-4504	-3599	-4425	-3606	-5245	-3770
	Absorbent irradiation effect ($\Delta\rho$ = 100 pcm)												
11	0	280	0	0	100	**-4231**	**-3555**	**-4404**	**-3499**	**-4325**	**-3506**	**-5145**	**-3670**

Table 3.15a. Control rods worth calculation. States description

V1. BOC	V2. BOC	V3. BOC	V1. EOC	V1. EOC	V1. EOC
S1 Wnom, Xe=Xe eq, t_{entry}=287°C, Cb burnup 100 % 5↓* 30 % 10↓	**S1** MCL, Xe=0, t_{entry}=280°C Cb crit 30% 10↓	**S1** MCL, Xe=Xe eq, t_{entry}=280°C Cb crit 30% 10↓	**S1** Wnom, Xe=Xe eq, t_{entry}=287°C Cb burnup 100 % 5↓ 30% 10↓	**S1** MCL, Xe=Xe eq, t_{entry}=280°C Cb crit 100 % 5↓ 30 % 10↓	**S1** MCL, Xe=0, t_{entry}=280°C Cb crit 100 % 5↓ 30 % 10↓
S2: the same but 100% 1-10↓	**S2**: the same but 100% 1-10↓	**S2**: the same but 100% 1-10↓	**S2**: the same but 100% 1-10↓	**S2**: the same but 100% 1-10↓	**S2**: the same but 100% 1-10↓

Table 3.15b. Control rods worth in Uranium reference core and in 3 MOX LTAs loaded cores (pcm)

		Uranium Core			MOX-1			MOX-2			MOX-3		
	Variant	V1	V2	V3	V1	V2	V3	V1	V2	V3	V1	V2	V3
BOC	Stuck rod number	55	55	55	67	67	67	109	82	82	112	97	97
	$(RO)_{AP}$	6930	6770	6730	6980	6830	6800	6960	6790	6730	7700	7150	7120
EOC	Stuck rod number	55	55	55	97	97	97	55	97	97	97	55	55
	$(RO)_{AP}$	7200	6150	6150	7100	6010	5990	7140	6090	6120	7170	6190	6170

* X% N↓ means that the Bank N is X% inserted in core

Table 3.16. Core reactivity in the process of control rods movement

AP Position,% (Hreg=80%)	BOC							
	Uranium		MOX-1		MOX-2		MOX-3	
	No stuck	Stuck N 55	No stuck	Stuck N 67	No stuck	Stuck N 109	No stuck	Stuck N 112
100	0	0	0	0	0	0	0	0
90	-120	-120	-120	-120	-120	-110	-120	-120
80	-210	-210	-210	-210	-200	-200	-210	-200
70	-310	-310	-310	-310	-300	-290	-300	-300
60	-460	-460	-450	-450	-430	-430	-440	-440
50	-700	-700	-690	-680	-660	-660	-680	-670
40	-1150	-1140	-1110	-1110	-1070	-1070	-1090	-1090
30	-2000	-1990	-1920	-1920	-1860	-1850	-1900	-1890
20	-3620	-3590	-3500	-3480	-3430	-3410	-3490	-3470
10	-7050	-6810	-6950	-6740	-6910	-6660	-7010	-6890
0	-9150	-8330	-9070	-8300	-9070	-8190	-9270	-8940

AP Position,% (Hreg=80%)	EOC							
	Uranium		MOX-1		MOX-2		MOX-3	
	No stuck	Stuck N 55	No stuck	Stuck N 97	No stuck	Stuck N 97	No stuck	Stuck N 97
100	0	0	0	0	0	0	0	0
90	-140	-140	-140	-140	-130	-130	-140	-140
80	-190	-190	-190	-190	-190	-190	-190	-190
70	-260	-260	-260	-250	-250	-250	-260	-260
60	-360	-360	-350	-350	-350	-350	-350	-350
50	-530	-530	-530	-530	-520	-520	-530	-520
40	-880	-870	-870	-860	-850	-850	-860	-860
30	-1590	-1580	-1570	-1560	-1530	-1530	-1550	-1540
20	-3000	-2980	-2950	-2930	-2900	-2890	-2920	-2900
10	-6300	-6160	-6190	-6050	-6170	-6020	-6200	-6060
0	-9410	-8570	-9310	-8480	-9320	-8440	-9400	-8560

Table 3.17. Return Criticality Temperature

	UOX	MOX-1	MOX-2	MOX-3
RCT, °C	124	128	128	117

Figure 2.1. Simplified Design for Uranium Reference Assembly (Type A)

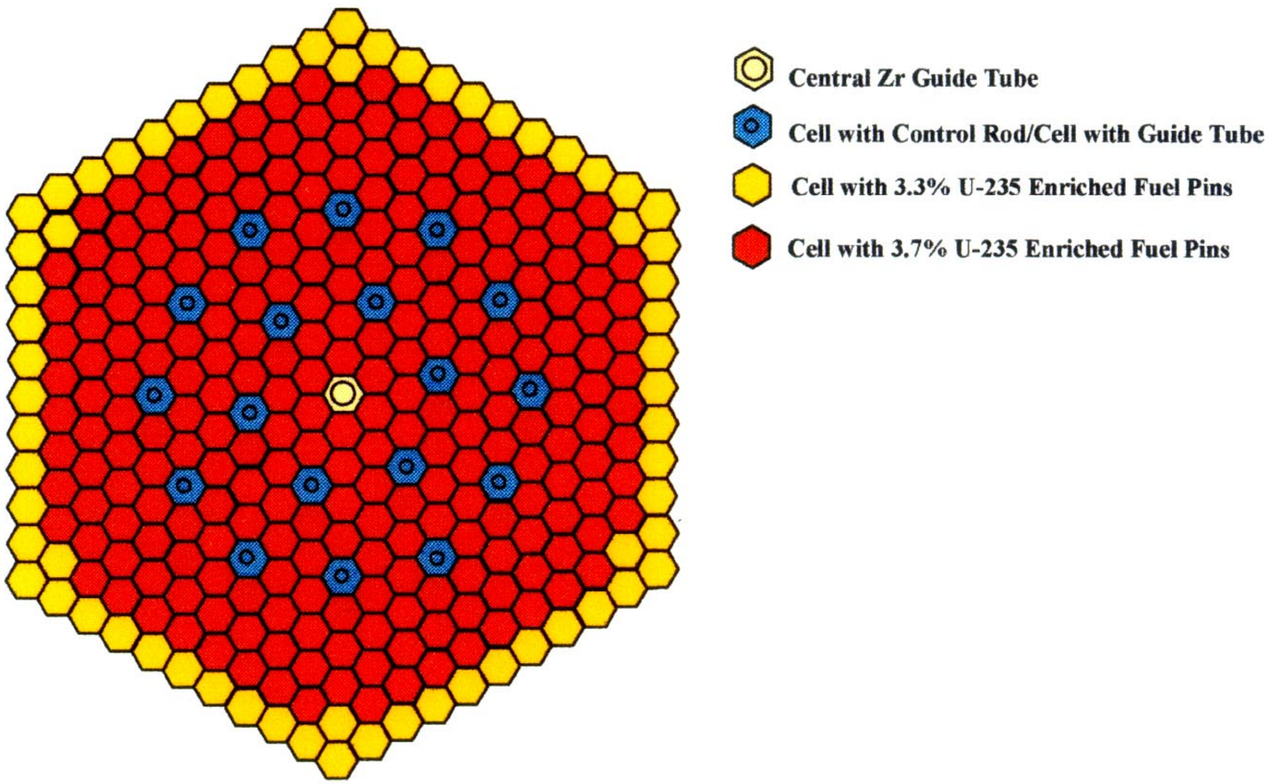


Figure 2.2. Calculational Model for Reference Uranium Assembly Surrounded by Uranium Assemblies. 60° Sector

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 absorber
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 2.3. Simplified Design for Uranium Assembly (Types B and Ba)

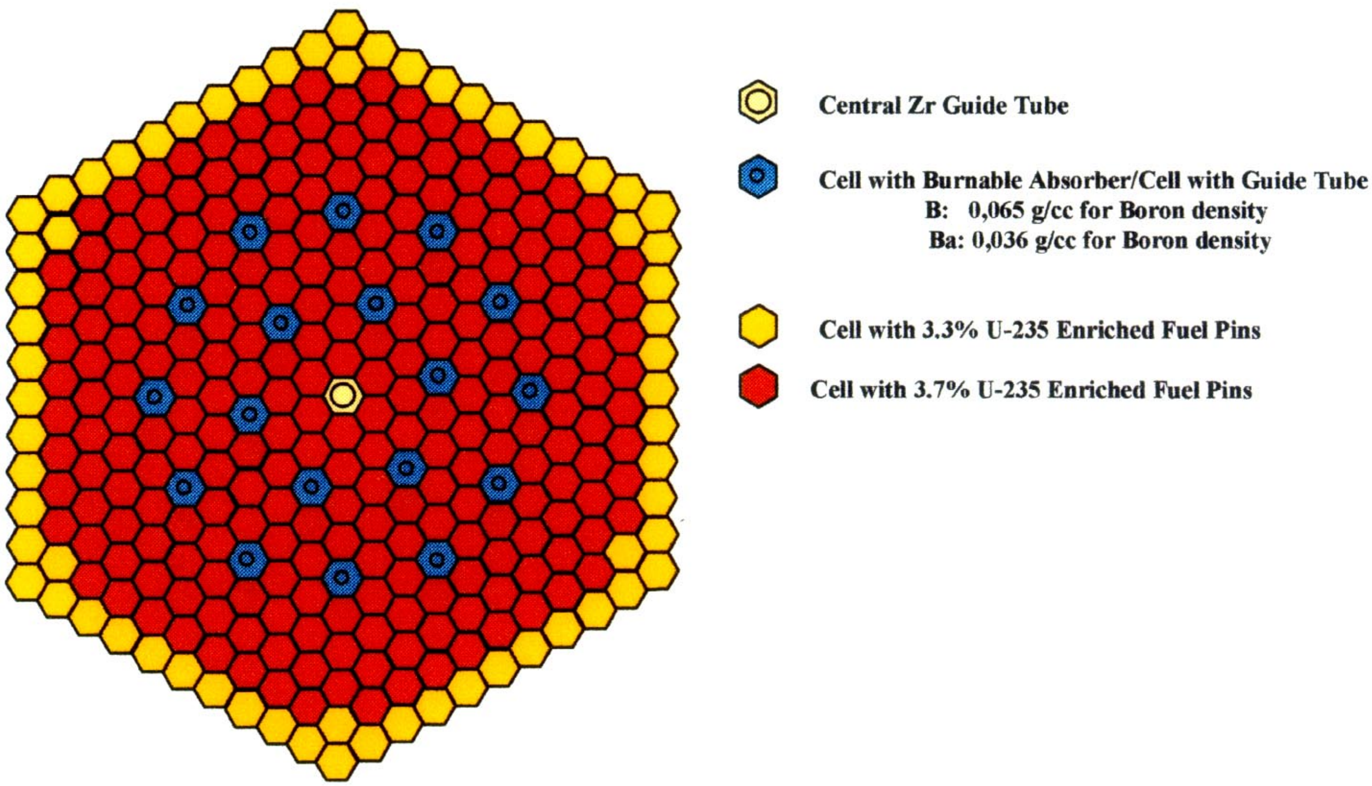


Figure 2.4. Simplified Design for Uranium Assembly (Type C)

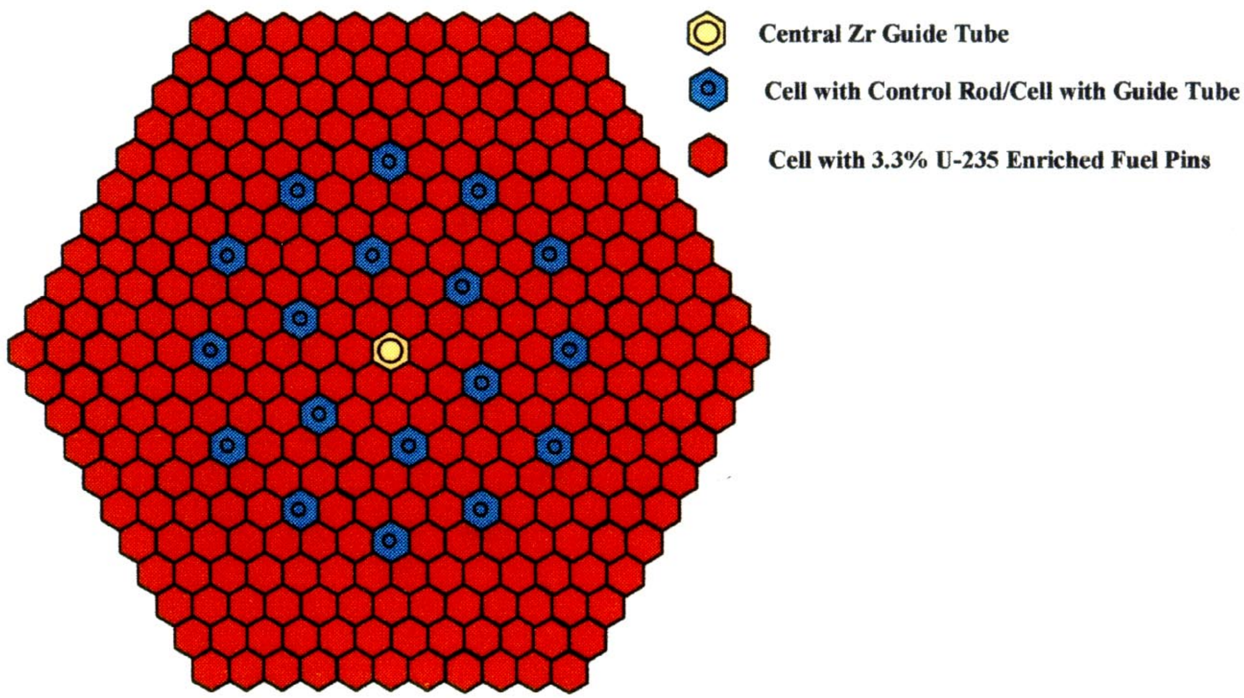


Figure 2.5. Simplified Design for 100 % Plutonium (3 Zones) MOX LTA

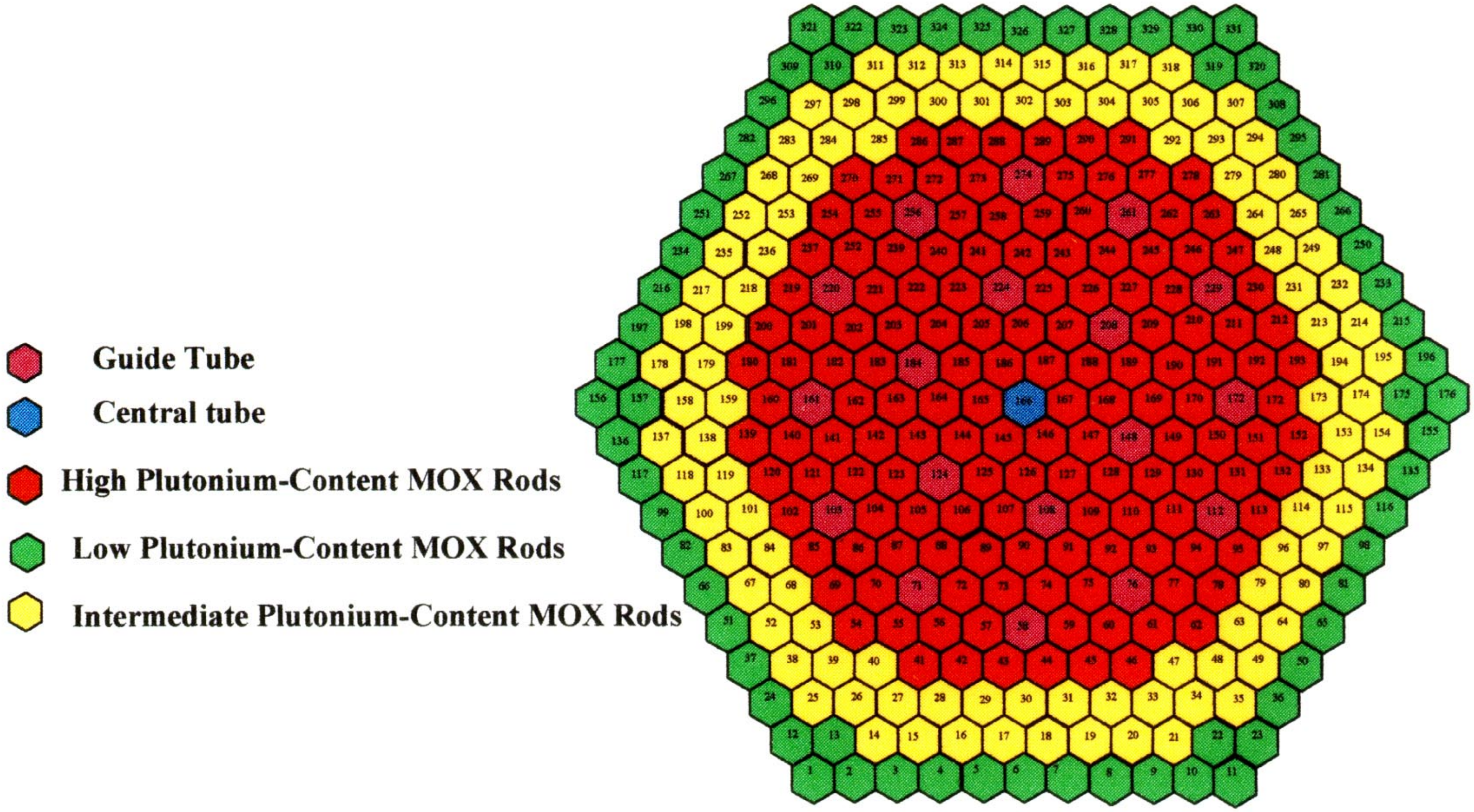
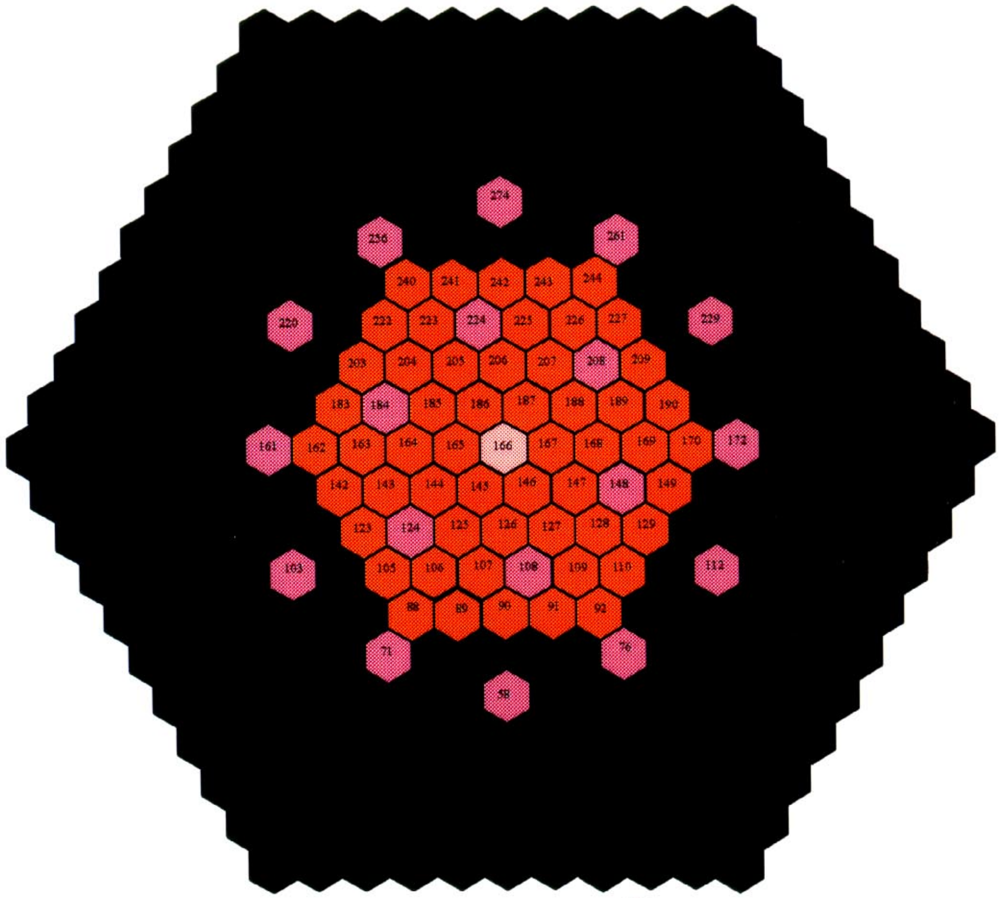


Figure 2.6. Calculational Model for 3-Zones (100 % Plutonium) MOX LTA Surrounded by Uranium Assemblies. 60° Sector

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,
71,71,29,71,71,71,29,71,71,71,71,25,64,57,57,
71,71,71,71,71,71,71,71,71,71,71,25,64,57,57,57,
29,71,71,71,71,29,71,71,71,71,71,25,64,57,57,50,50,
71,71,71,29,71,71,71,71,71,71,71,25,64,57,57,50,50,29,
71,71,71,71,71,71,71,71,71,71,71,25,64,57,57,50,50,50,50,
71,71,71,71,71,71,71,71,71,71,71,25,64,57,57,50,29,50,50,50,
71,71,71,71,71,71,71,71,71,71,71,25,64,57,57,50,50,50,50,50,50,
71,71,71,71,71,71,71,71,71,71,71,25,64,57,57,50,50,50,50,29,50,50,
26,25,25,25,25,25,25,25,25,25,25,26,64,64,57,57,50,29,50,50,50,50,27,

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

Figure 2.7. Simplified Design for "Island-1" Type MOX LTA



Enriched Uranium Rods

High Plutonium-Content MOX Rods

Central tube

Guide Tube

Figure 2.8. 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,
71,71,29,71,71,71,29,71,71,71,71,25,64,57,57,
71,71,71,71,71,71,71,71,71,71,71,25,64,57,57,57,
29,71,71,71,71,29,71,71,71,71,71,25,64,57,57,57,57,
71,71,71,29,71,71,71,71,71,71,71,25,64,57,57,57,57,28,
71,71,71,71,71,71,71,71,71,71,71,25,64,57,57,57,57,57,50,
71,71,71,71,71,71,71,71,71,71,71,25,64,57,57,57,28,57,50,50,
71,71,71,71,71,71,71,71,71,71,71,25,64,57,57,57,57,57,50,50,50,
71,71,71,71,71,71,71,71,71,71,71,25,64,57,57,57,57,57,50,28,50,50,
26,25,25,25,25,25,25,25,25,25,25,26,64,64,57,57,57,28,50,50,50,50,27,

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

Figure 2.9. Simplified Design for "Island-2" Type MOX LTA

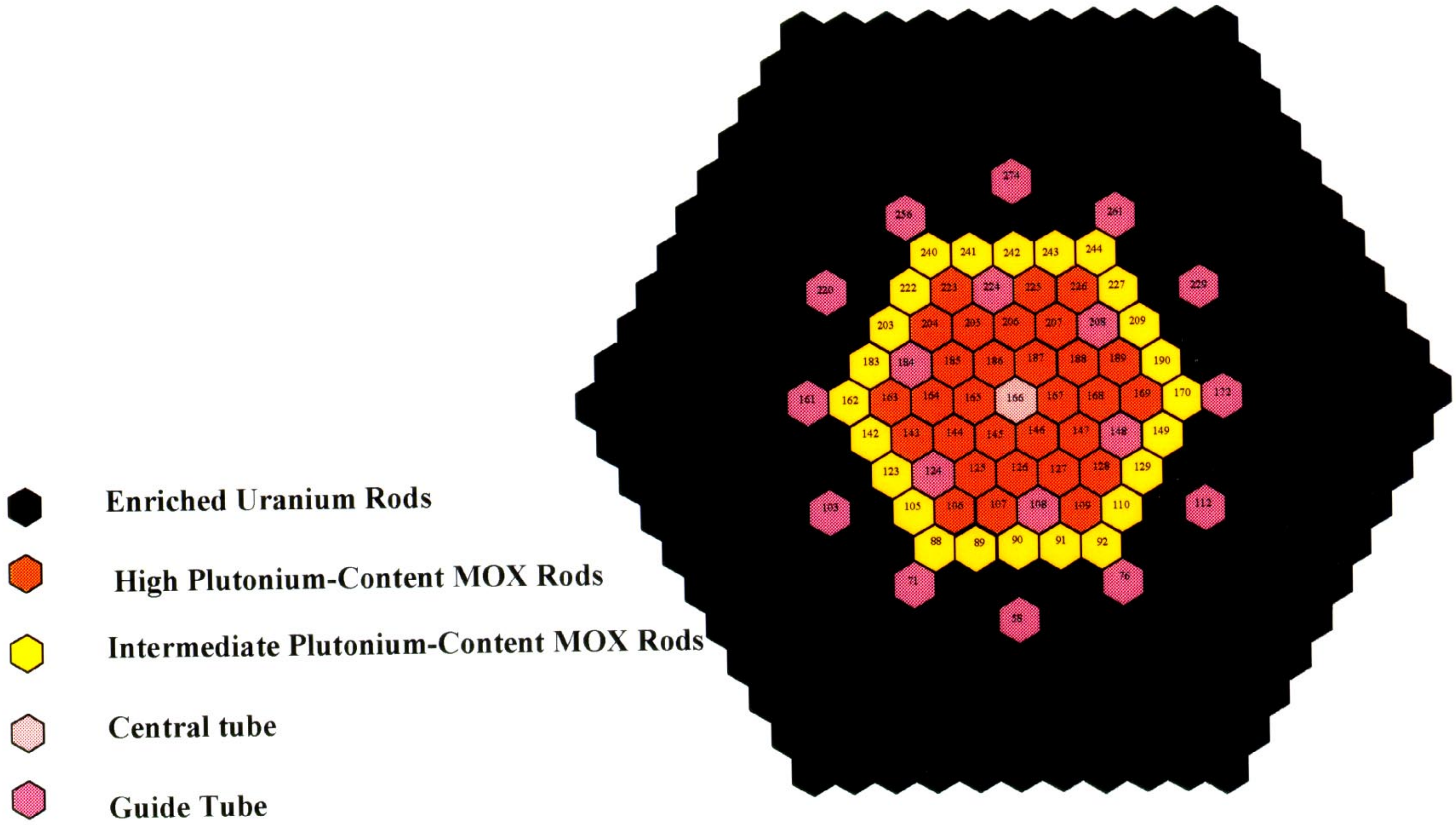


Figure 2.10. Calculational Model for "Island-2" MOX LTA Surrounded by Uranium Assemblies. 60° Sector

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,
71,71,71,29,71,71,71,71,71,71,71,25,57,
71,71,71,71,71,71,71,71,71,71,71,25,57,57,
71,71,29,71,71,71,29,71,71,71,71,25,57,57,57,
71,71,71,71,71,71,71,71,71,71,71,25,57,57,57,57,
29,71,71,71,71,29,71,71,71,71,71,25,57,57,57,57,57,
71,71,71,29,71,71,71,71,71,71,71,25,57,57,57,57,57,28,
71,71,71,71,71,71,71,71,71,71,71,25,57,57,57,57,57,57,64,
71,71,71,71,71,71,71,71,71,71,71,25,57,57,57,57,28,57,64,64,
71,71,71,71,71,71,71,71,71,71,71,25,57,57,57,57,57,57,64,50,50,
71,71,71,71,71,71,71,71,71,71,71,25,57,57,57,57,57,57,64,28,50,50,
26,25,25,25,25,25,25,25,25,25,25,26,57,57,57,57,57,28,64,64,50,50,27,

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

Figure 2.11. 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 absorber

223 –plutonium fuel rods

71 – uranium 3.7% U-235 fuel rods

Figure 2.12. Evolution of Ko in Plutonium-Uranium Super-Cells

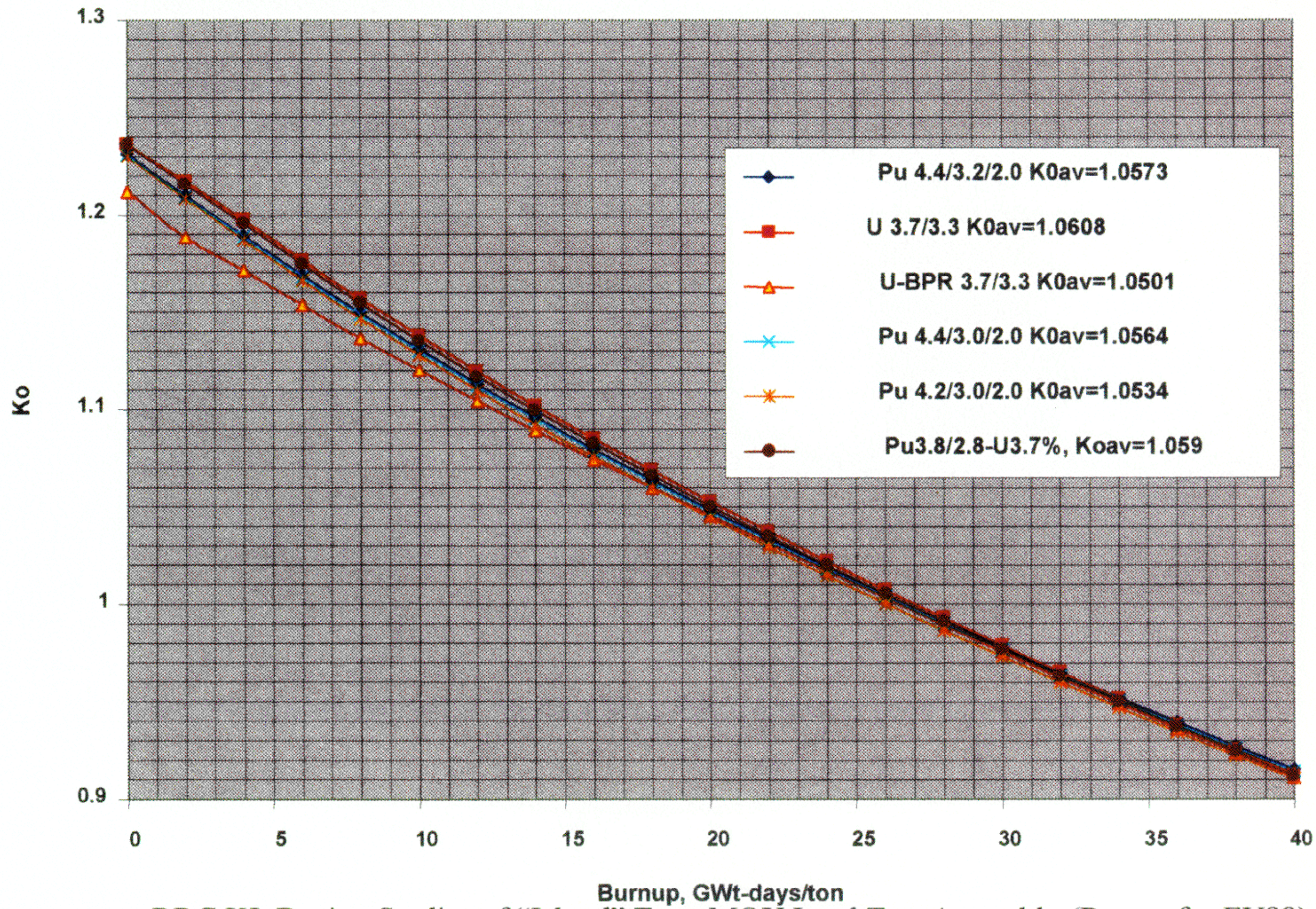


Figure 2.13. Evolution of Kk in Plutonium-Uranium Super-Cells

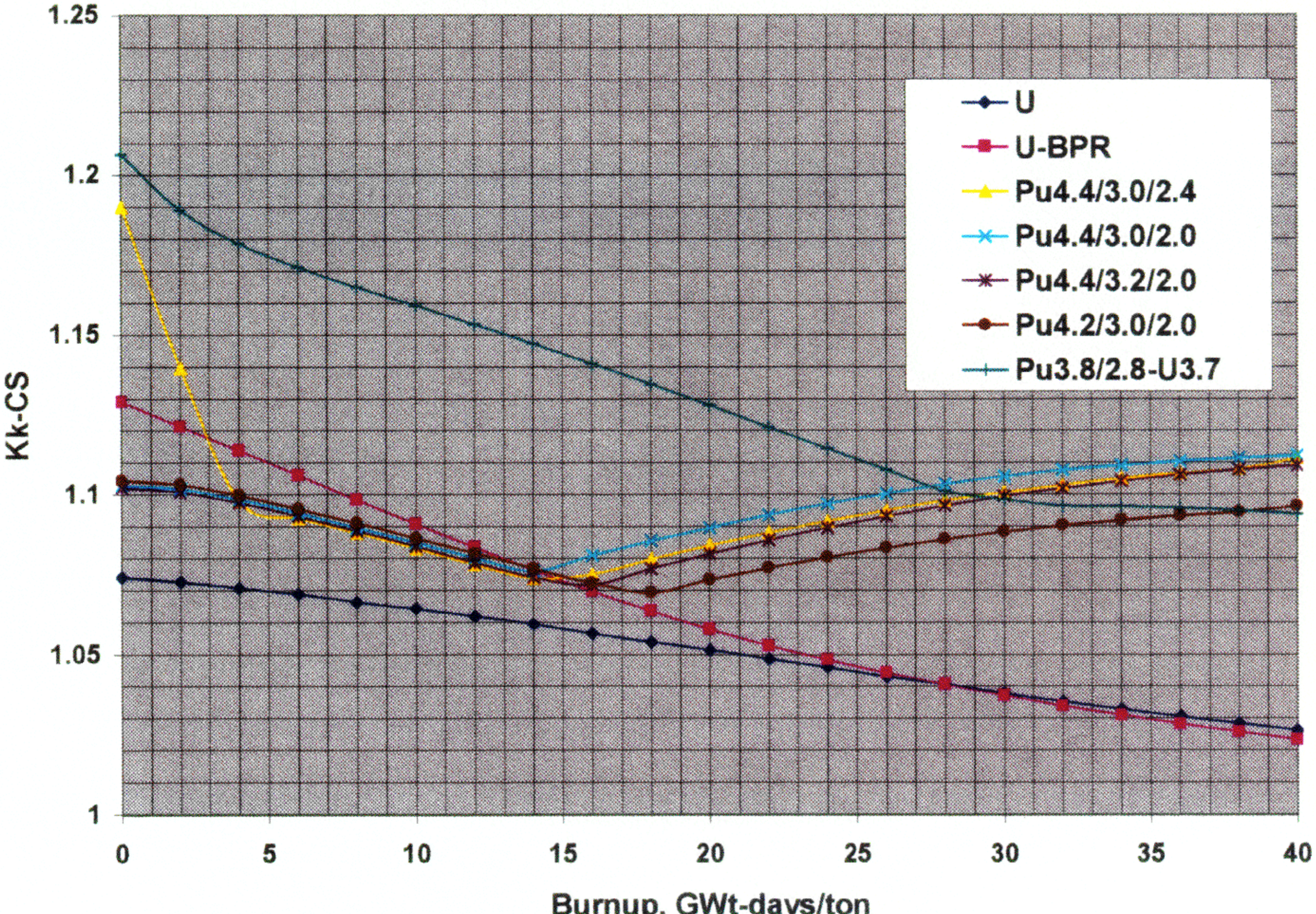
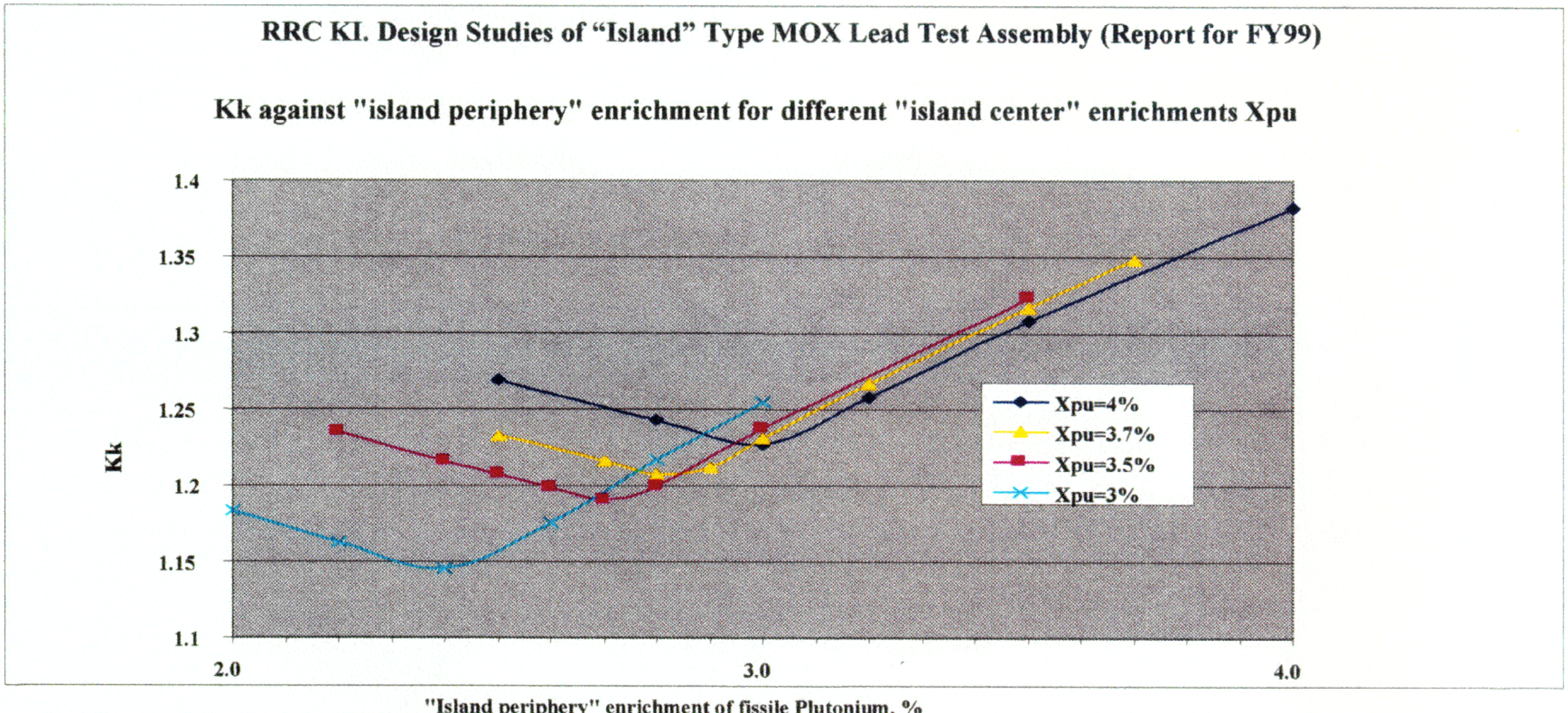
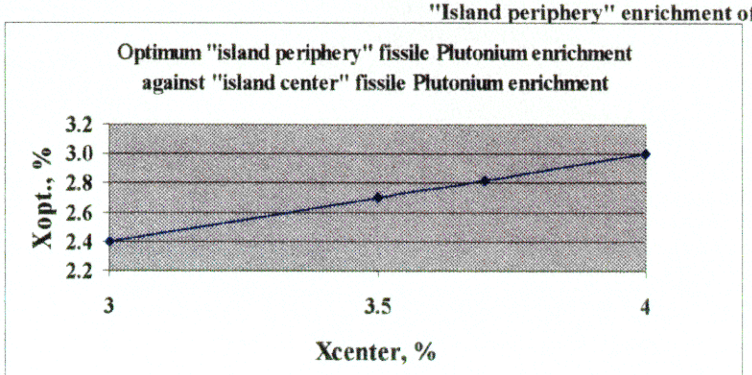


Figure 2.14. Parametric Studies of «Island» Type MOX LTA (U 3.7%)





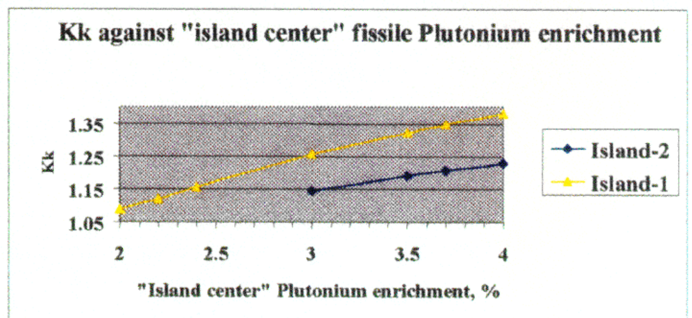
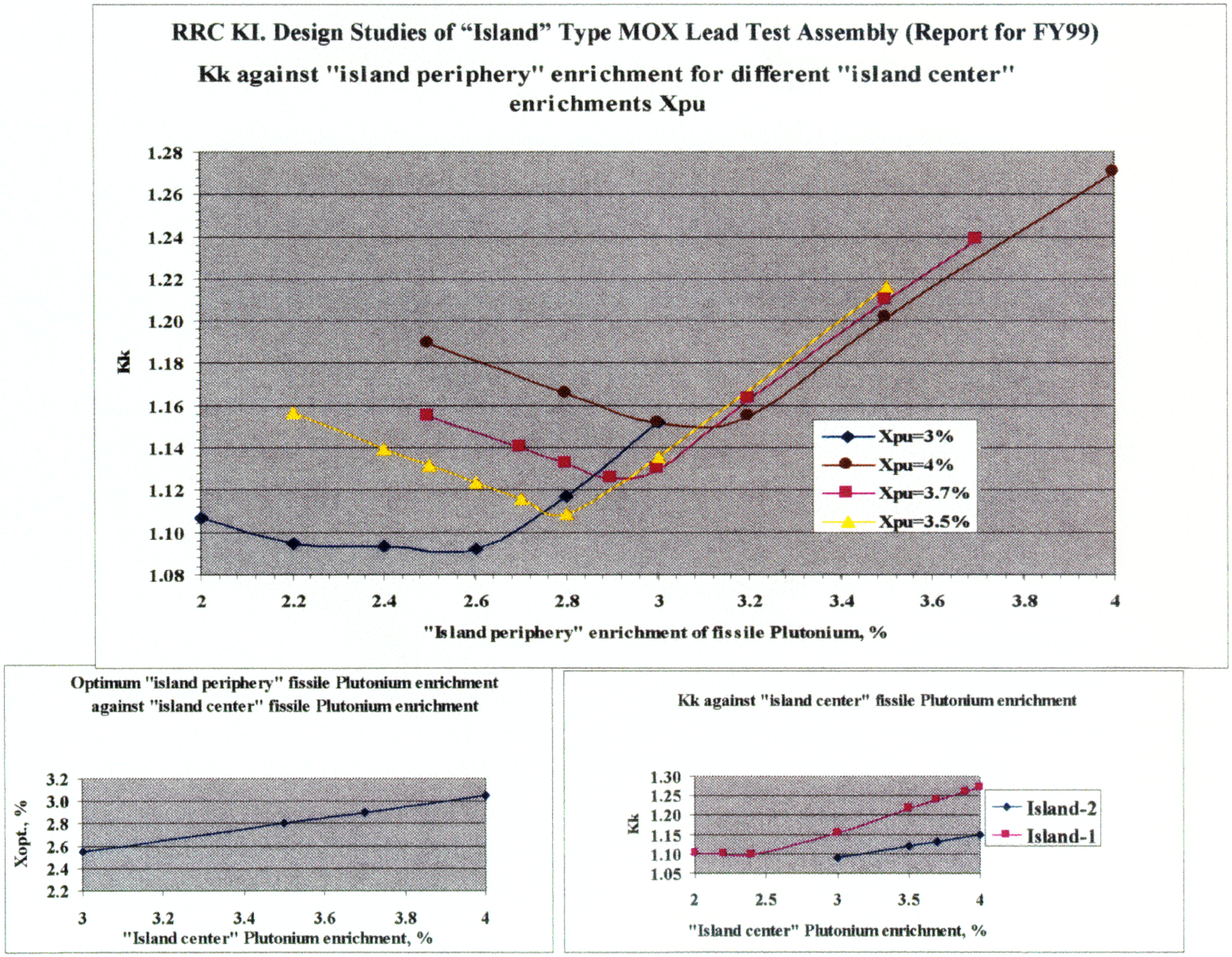


Figure 2.15. Parametric Studies of «Island» Type MOX LTA (U 4.4%)



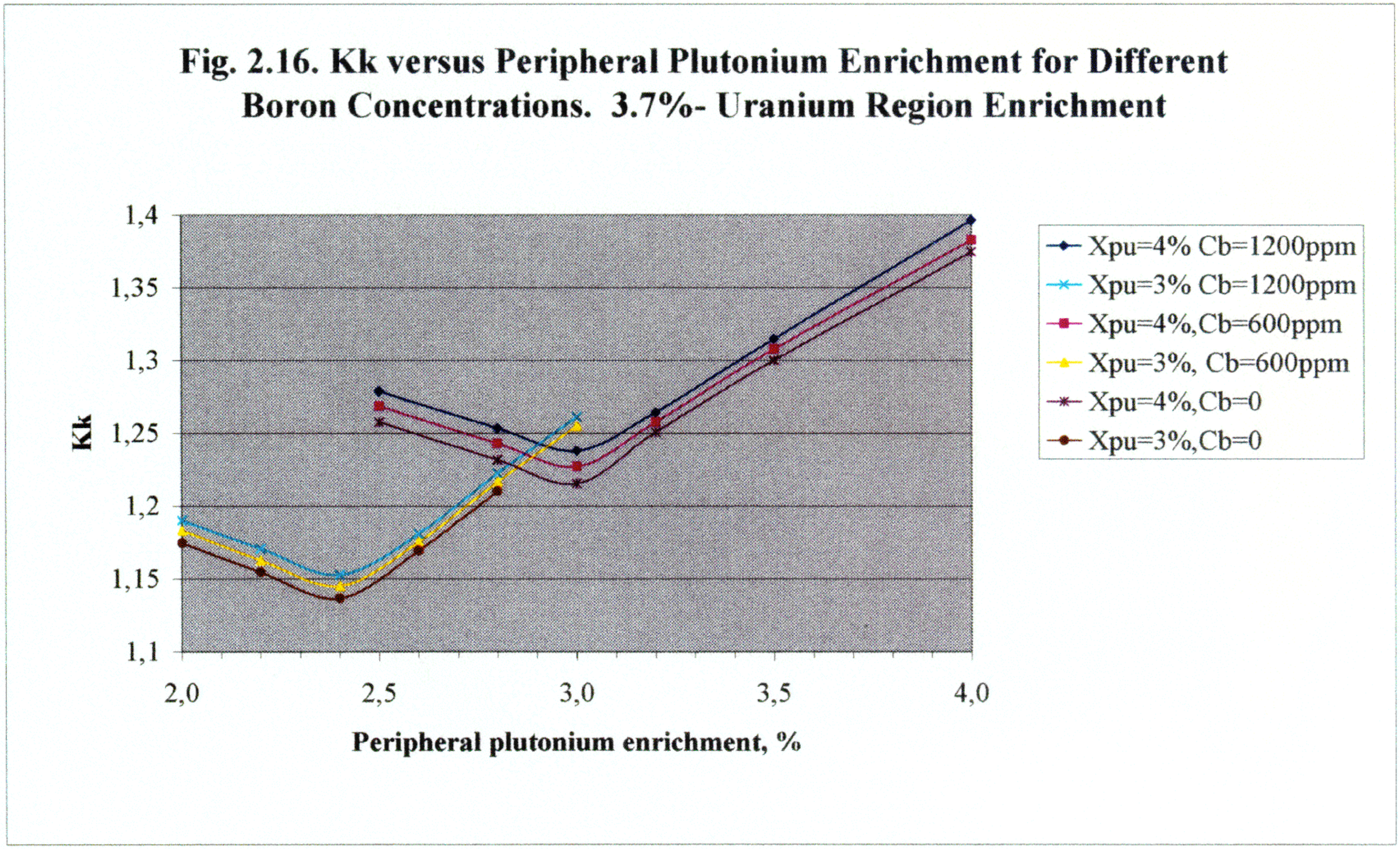


Fig. 2.16. Kk versus Peripheral Plutonium Enrichment for Different Boron Concentrations. 3.7%- Uranium Region Enrichment
Kk
1,4
1,35
1,3
1,25
1,2
1,15
1,1
2,0
2,5
3,0
3,5
4,0
Peripheral plutonium enrichment, %
Xpu=4% Cb=1200ppm
Xpu=3% Cb=1200ppm
Xpu=4%,Cb=600ppm
Xpu=3%, Cb=600ppm
Xpu=4%,Cb=0
Xpu=3%,Cb=0

Fig. 2.17. Kk versus Peripheral Plutonium Enrichment for Different Boron Concentrations. 4.4%- Uranium Region Enrichment

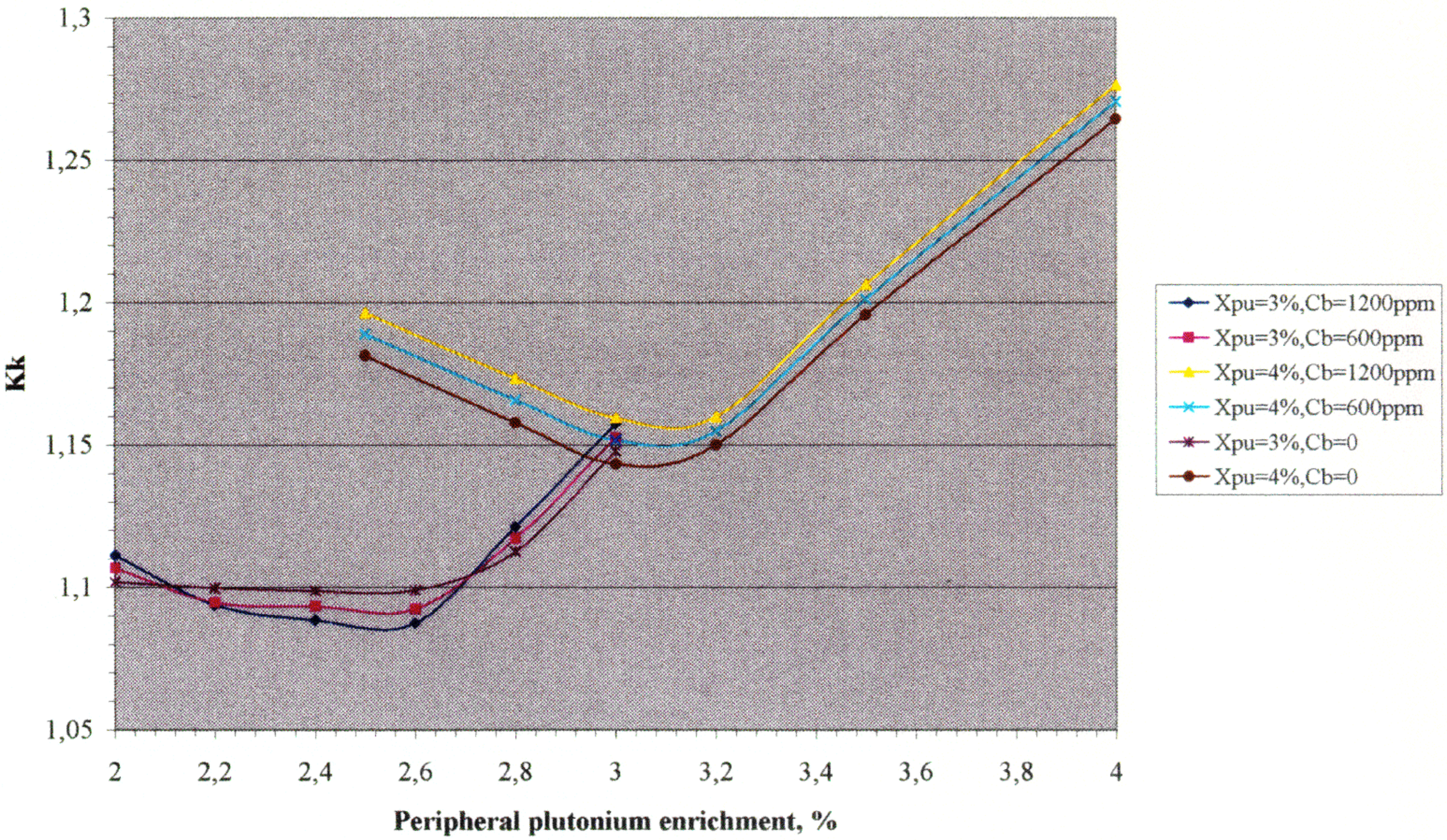
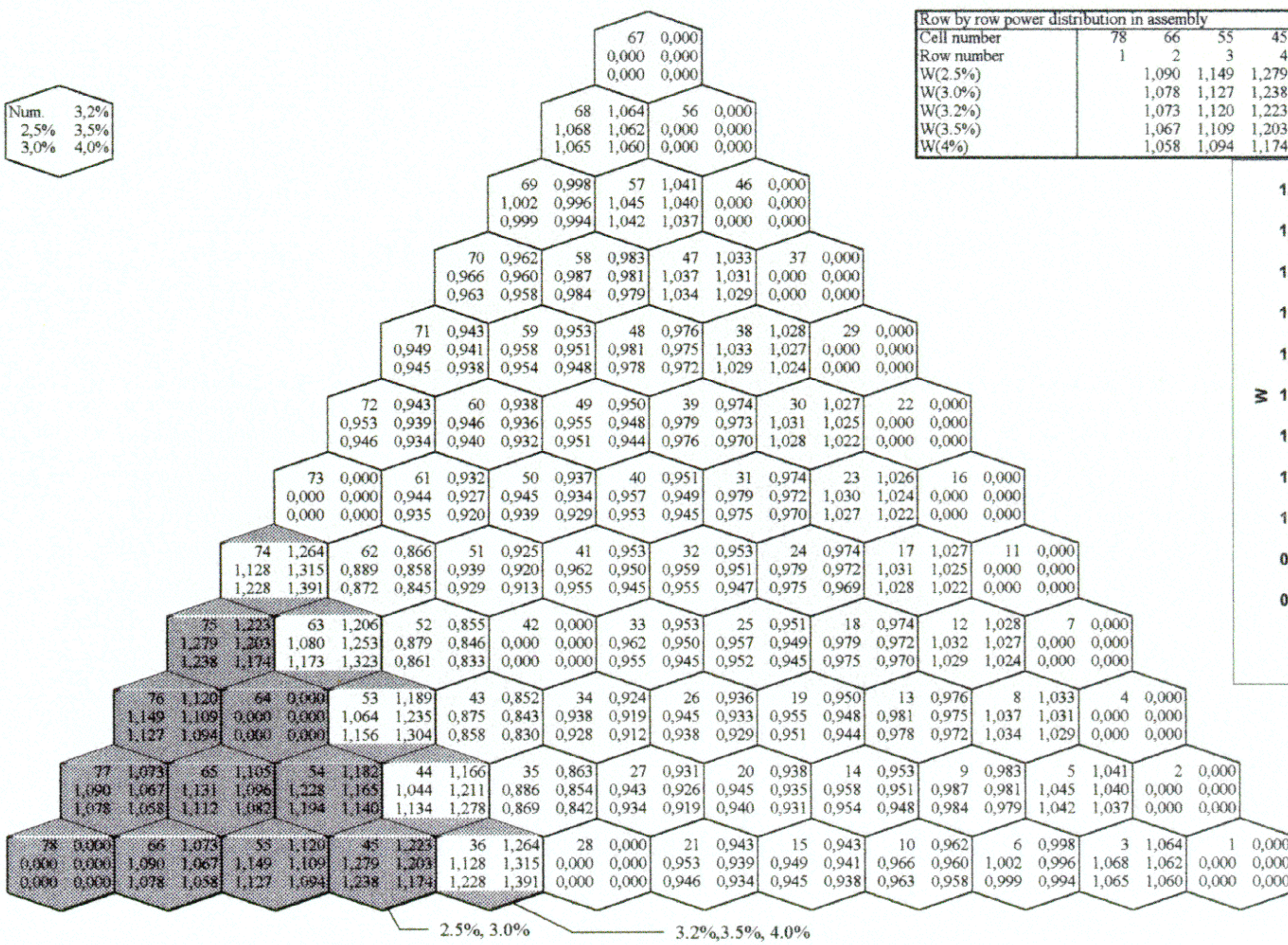


Fig. 2.18. Inter - assembly Power Distributions versus Peripheral Plutonium Enrichments. 4%- Central Plutonium Enrichment. 3.7%- Uranium Region Enrichment. Cb(nat)=1200ppm



Row by row power distribution in assembly											
Cell number	78	66	55	45	36	28	21	15	10	6	3
Row number	1	2	3	4	5	6	7	8	9	10	11
W(2.5%)		1,090	1,149	1,279	1,128		0,953	0,949	0,966	1,002	1,068
W(3.0%)		1,078	1,127	1,238	1,228		0,946	0,945	0,963	0,999	1,065
W(3.2%)		1,073	1,120	1,223	1,264		0,943	0,943	0,962	0,998	1,064
W(3.5%)		1,067	1,109	1,203	1,315		0,939	0,941	0,960	0,996	1,062
W(4%)		1,058	1,094	1,174	1,391		0,934	0,938	0,958	0,994	1,060

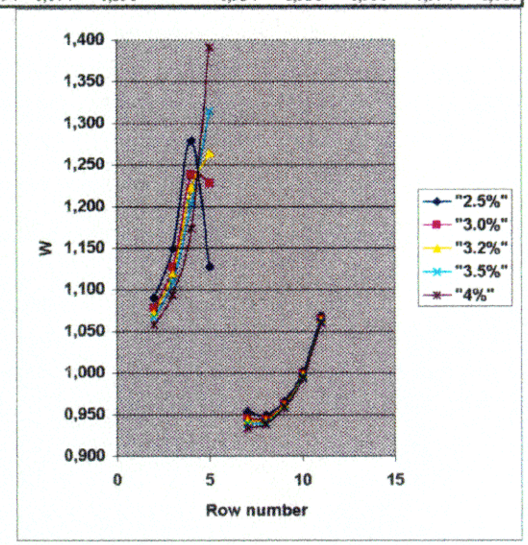
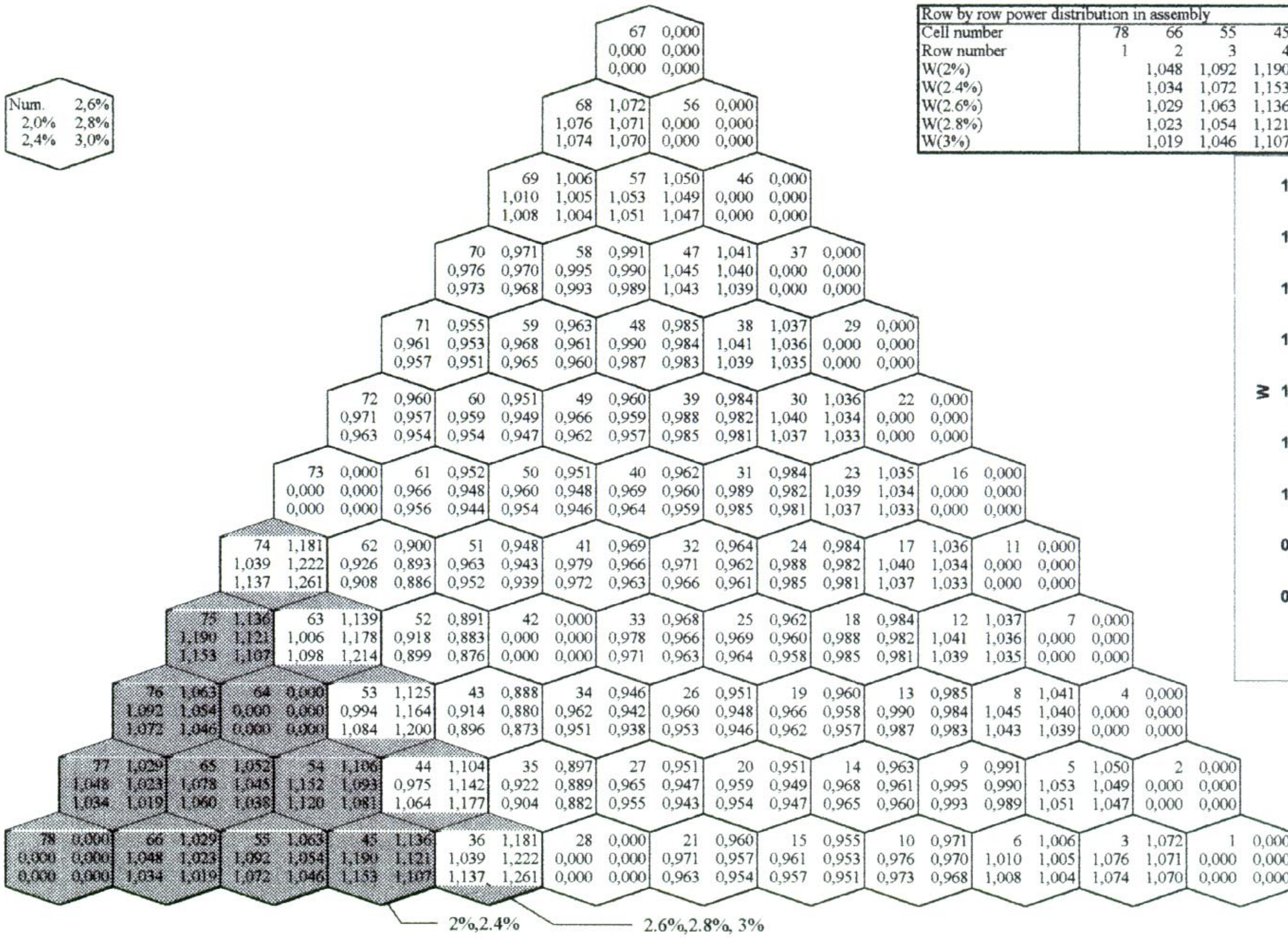


Fig. 2.19. Inter - assembly Power Distributions versus Peripheral Plutonium Enrichments. 3%- Central Plutonium Enrichment. 3.7%- Uranium Region Enricnment. Cb(nat)=1200ppm



Row by row power distribution in assembly											
Cell number	78	66	55	45	36	28	21	15	10	6	3
Row number	1	2	3	4	5	6	7	8	9	10	11
W(2%)		1,048	1,092	1,190	1,039		0,971	0,961	0,976	1,010	1,076
W(2.4%)		1,034	1,072	1,153	1,137		0,963	0,957	0,973	1,008	1,074
W(2.6%)		1,029	1,063	1,136	1,181		0,960	0,955	0,971	1,006	1,072
W(2.8%)		1,023	1,054	1,121	1,222		0,957	0,953	0,970	1,005	1,071
W(3%)		1,019	1,046	1,107	1,261		0,954	0,951	0,968	1,004	1,070

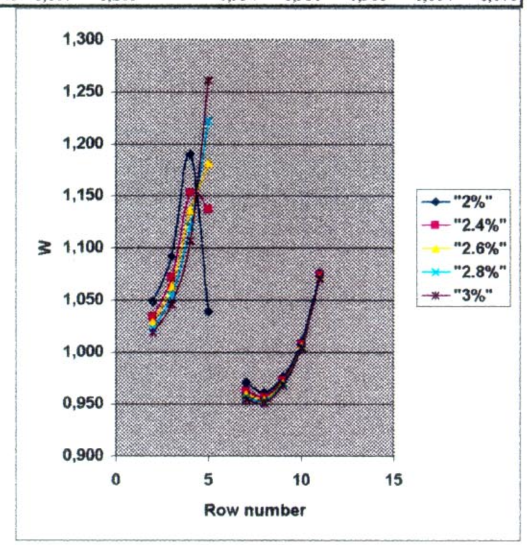
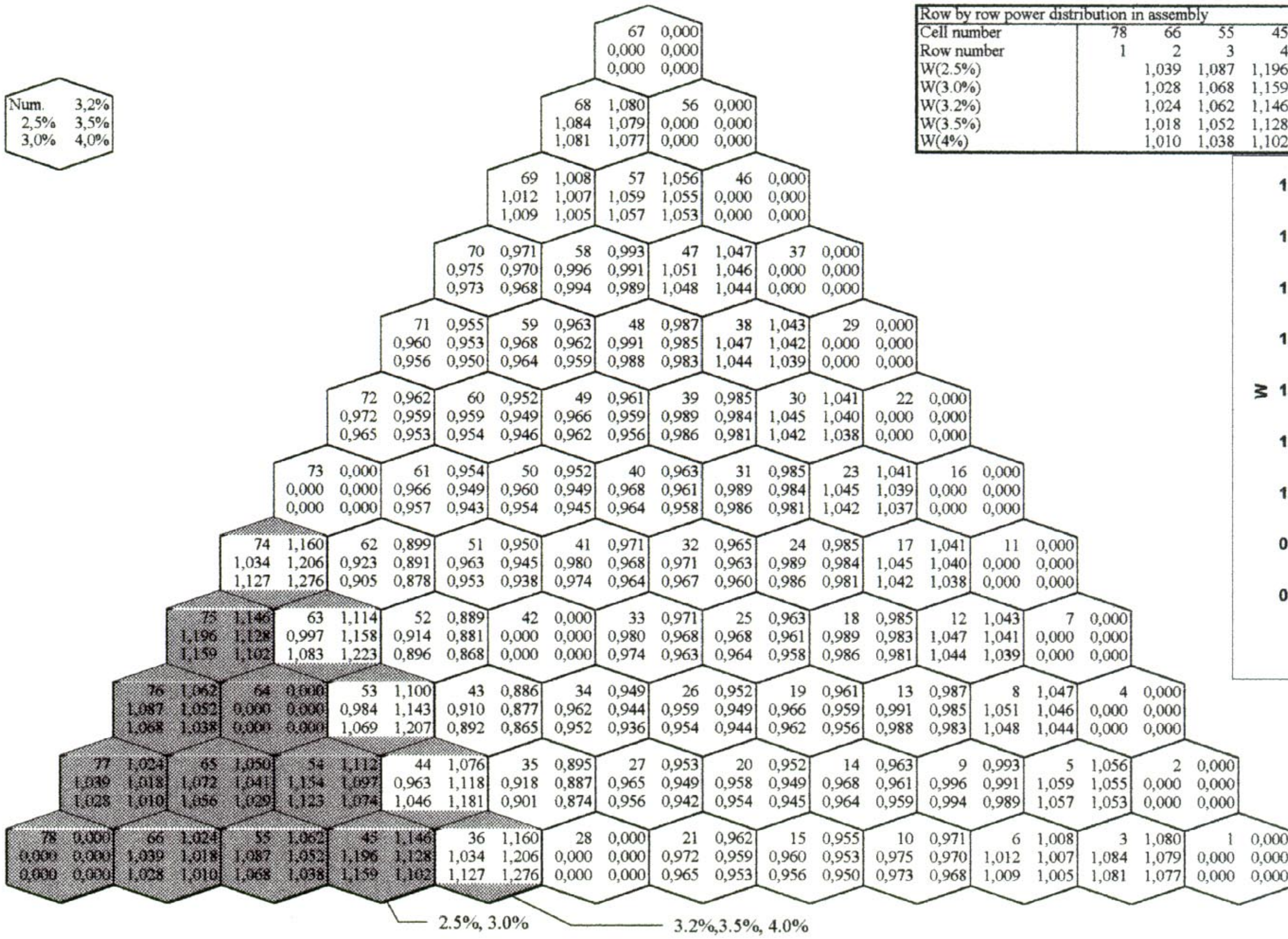


Fig. 2.20. Inter - assembly Power Distributions versus Peripheral Plutonium Enrichments. 4%- Central Plutonium Enrichment. 4.4%- Uranium Region Enricnment. Cb(nat)=1200ppm



Row by row power distribution in assembly											
Cell number	78	66	55	45	36	28	21	15	10	6	3
Row number	1	2	3	4	5	6	7	8	9	10	11
W(2.5%)		1,039	1,087	1,196	1,034		0,972	0,960	0,975	1,012	1,084
W(3.0%)		1,028	1,068	1,159	1,127		0,965	0,956	0,973	1,009	1,081
W(3.2%)		1,024	1,062	1,146	1,160		0,962	0,955	0,971	1,008	1,080
W(3.5%)		1,018	1,052	1,128	1,206		0,959	0,953	0,970	1,007	1,079
W(4%)		1,010	1,038	1,102	1,276		0,953	0,950	0,968	1,005	1,077

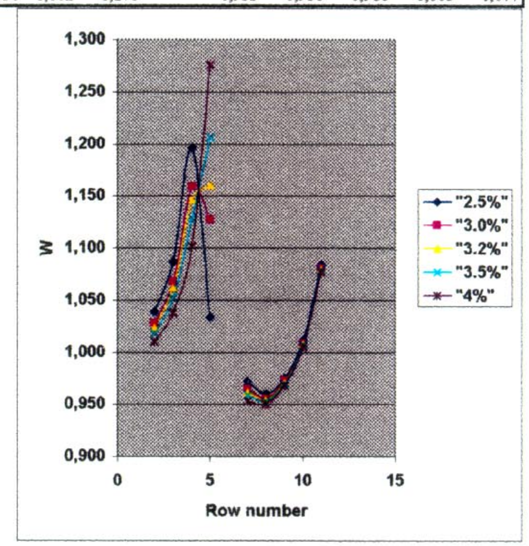
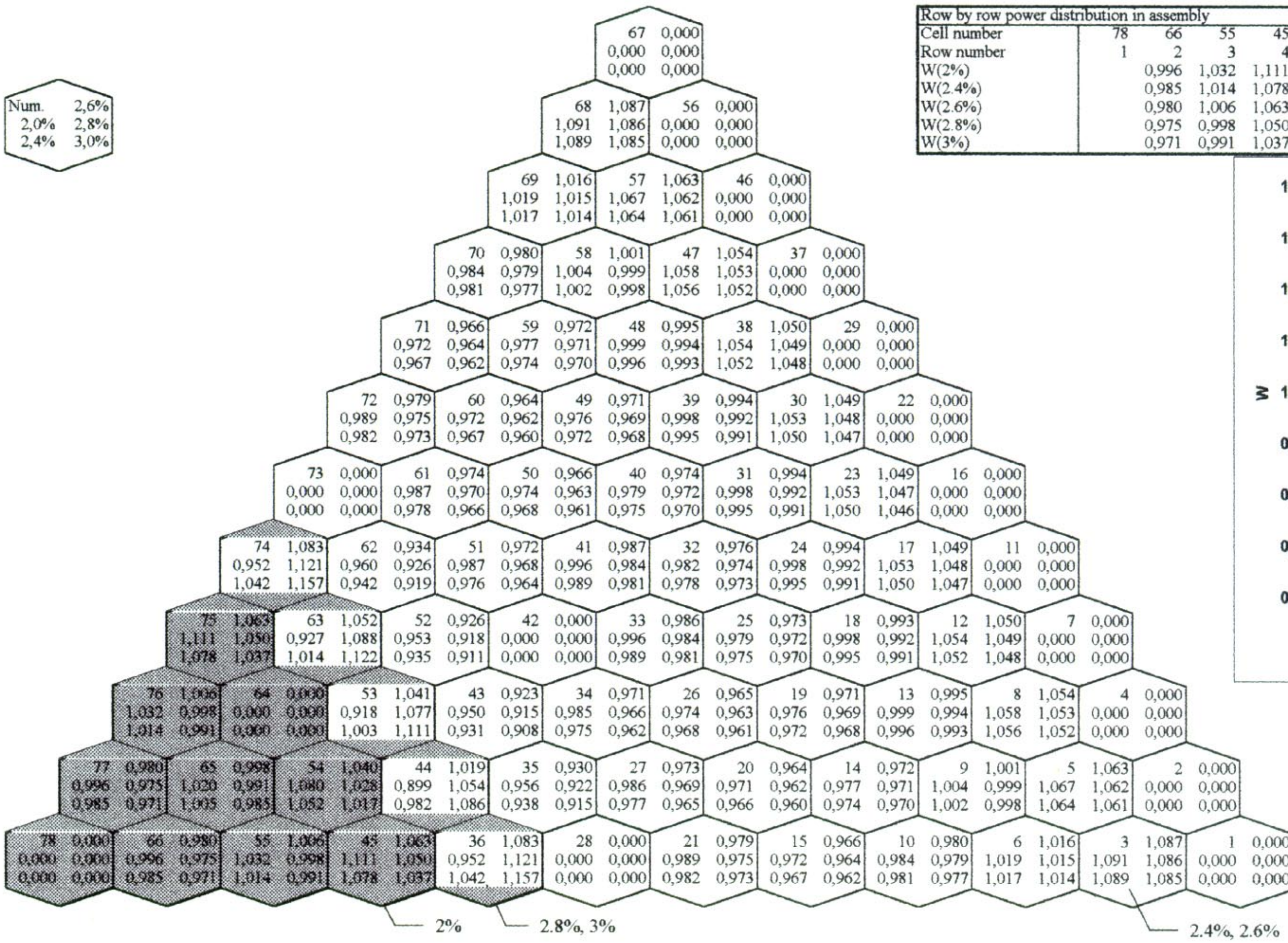


Fig. 2.21. Inter - assembly Power Distributions versus Peripheral Plutonium Enrichments. 3%- Central Plutonium Enrichment. 4.4%- Uranium Region Enricnment. Cb(nat)=1200ppm



Row by row power distribution in assembly											
Cell number	78	66	55	45	36	28	21	15	10	6	3
Row number	1	2	3	4	5	6	7	8	9	10	11
W(2%)		0,996	1,032	1,111	0,952		0,989	0,972	0,984	1,019	1,091
W(2.4%)		0,985	1,014	1,078	1,042		0,982	0,967	0,981	1,017	1,089
W(2.6%)		0,980	1,006	1,063	1,083		0,979	0,966	0,980	1,016	1,087
W(2.8%)		0,975	0,998	1,050	1,121		0,975	0,964	0,979	1,015	1,086
W(3%)		0,971	0,991	1,037	1,157		0,973	0,962	0,977	1,014	1,085

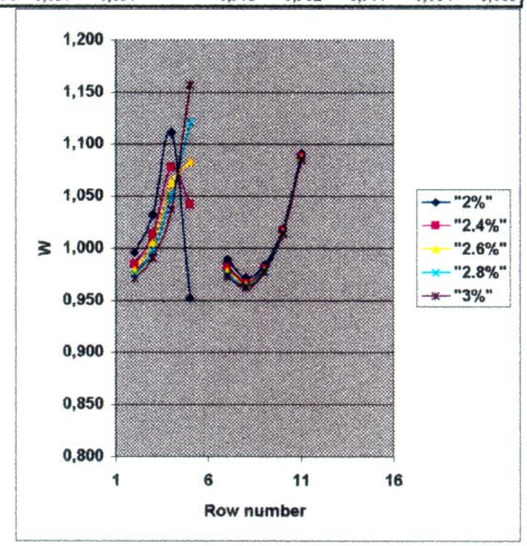
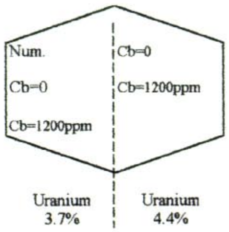
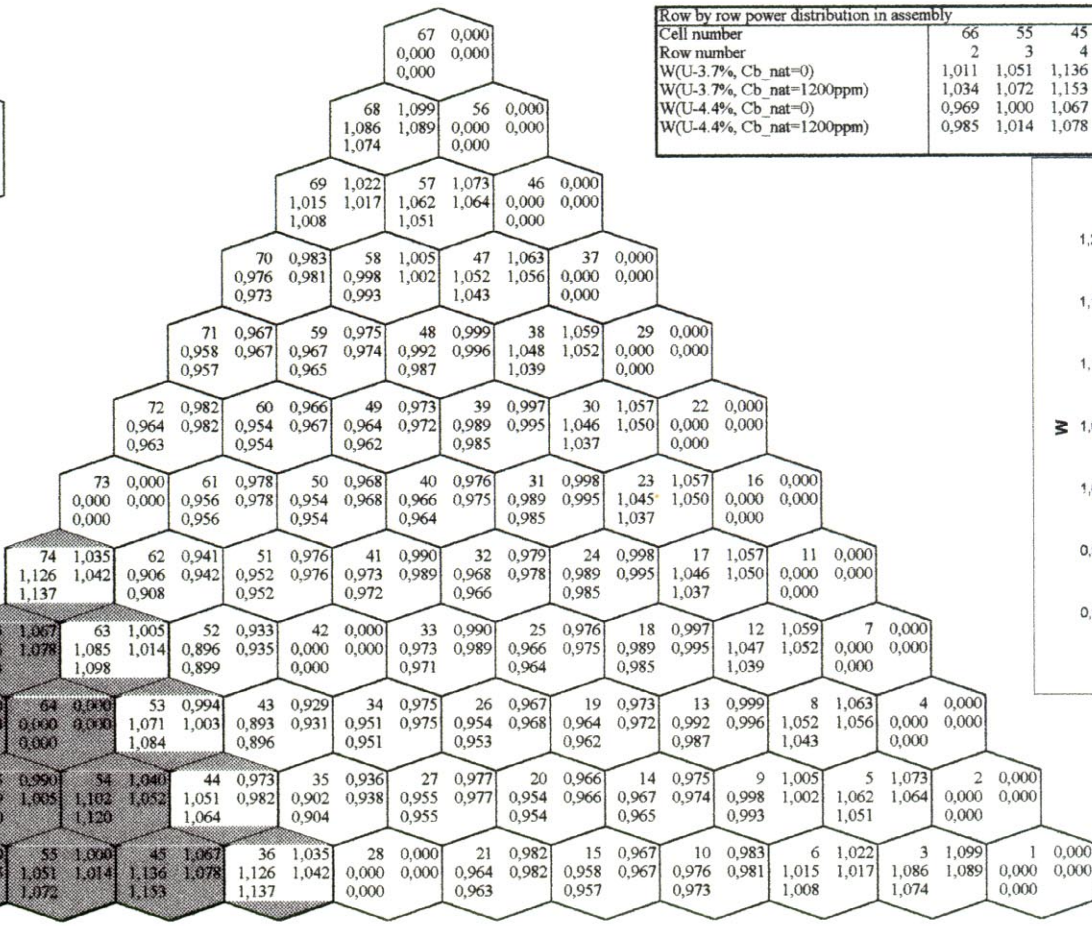


Fig. 2.22. Comparison of Power Inter-assembly Distributions in "Island-2" of Optimum Grading. 3% Plutonium Central Part with 3.7% and 4.4% Uranium Regions





Row by row power distribution in assembly										
Cell number	66	55	45	36	28	21	15	10	6	3
Row number	2	3	4	5	6	7	8	9	10	11
W(U-3.7%, Cb_nat=0)	1,011	1,051	1,136	1,126		0,964	0,958	0,976	1,015	1,086
W(U-3.7%, Cb_nat=1200ppm)	1,034	1,072	1,153	1,137		0,963	0,957	0,973	1,008	1,074
W(U-4.4%, Cb_nat=0)	0,969	1,000	1,067	1,035		0,982	0,967	0,983	1,022	1,099
W(U-4.4%, Cb_nat=1200ppm)	0,985	1,014	1,078	1,042		0,982	0,967	0,981	1,017	1,089

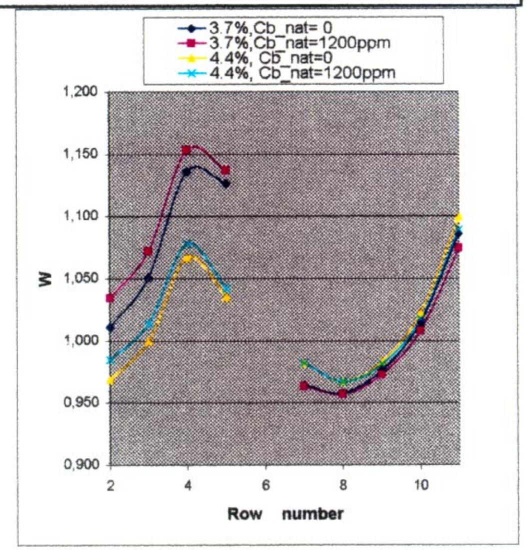
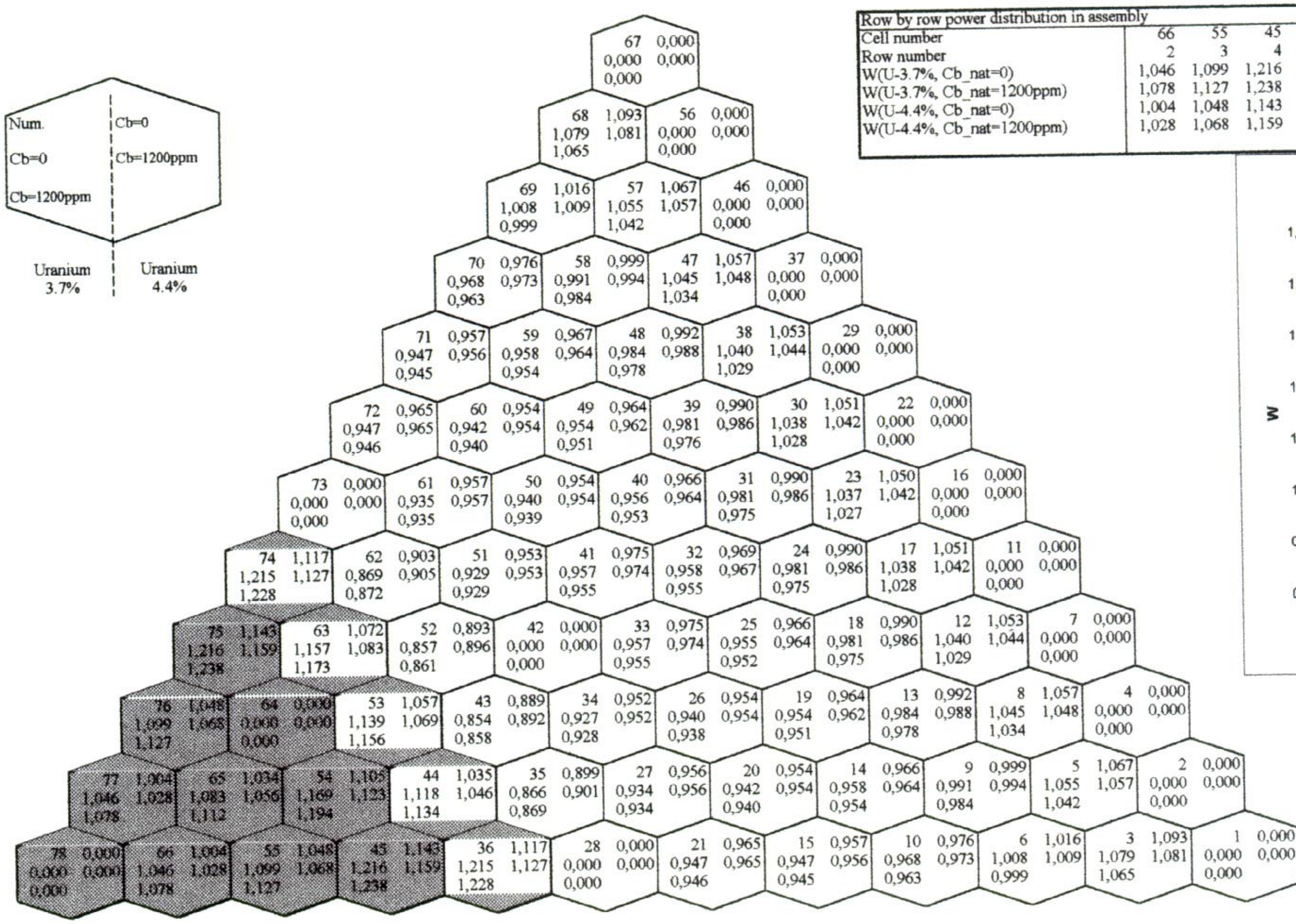


Fig. 2.23. Comparison of Power Inter-assembly Distributions in "Island-2" of Optimum Grading. 4% Plutonium Central Part with 3.7% and 4.4% Uranium Regions



Row by row power distribution in assembly										
Cell number	66	55	45	36	28	21	15	10	6	3
Row number	2	3	4	5	6	7	8	9	10	11
W(U-3.7%, Cb_nat=0)	1,046	1,099	1,216	1,215		0,947	0,947	0,968	1,008	1,079
W(U-3.7%, Cb_nat=1200ppm)	1,078	1,127	1,238	1,228		0,946	0,945	0,963	0,999	1,065
W(U-4.4%, Cb_nat=0)	1,004	1,048	1,143	1,117		0,965	0,957	0,976	1,016	1,093
W(U-4.4%, Cb_nat=1200ppm)	1,028	1,068	1,159	1,127		0,965	0,956	0,973	1,009	1,081

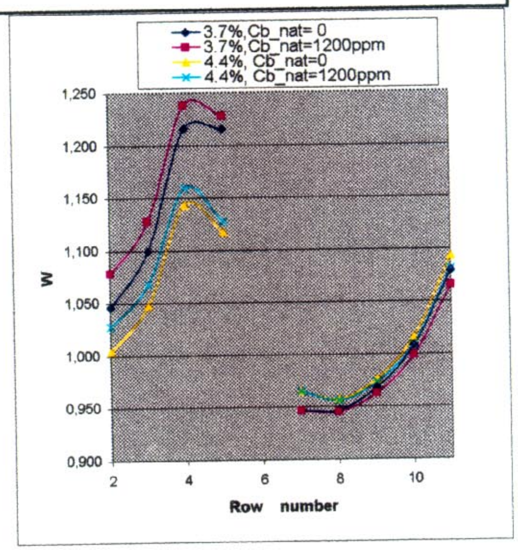
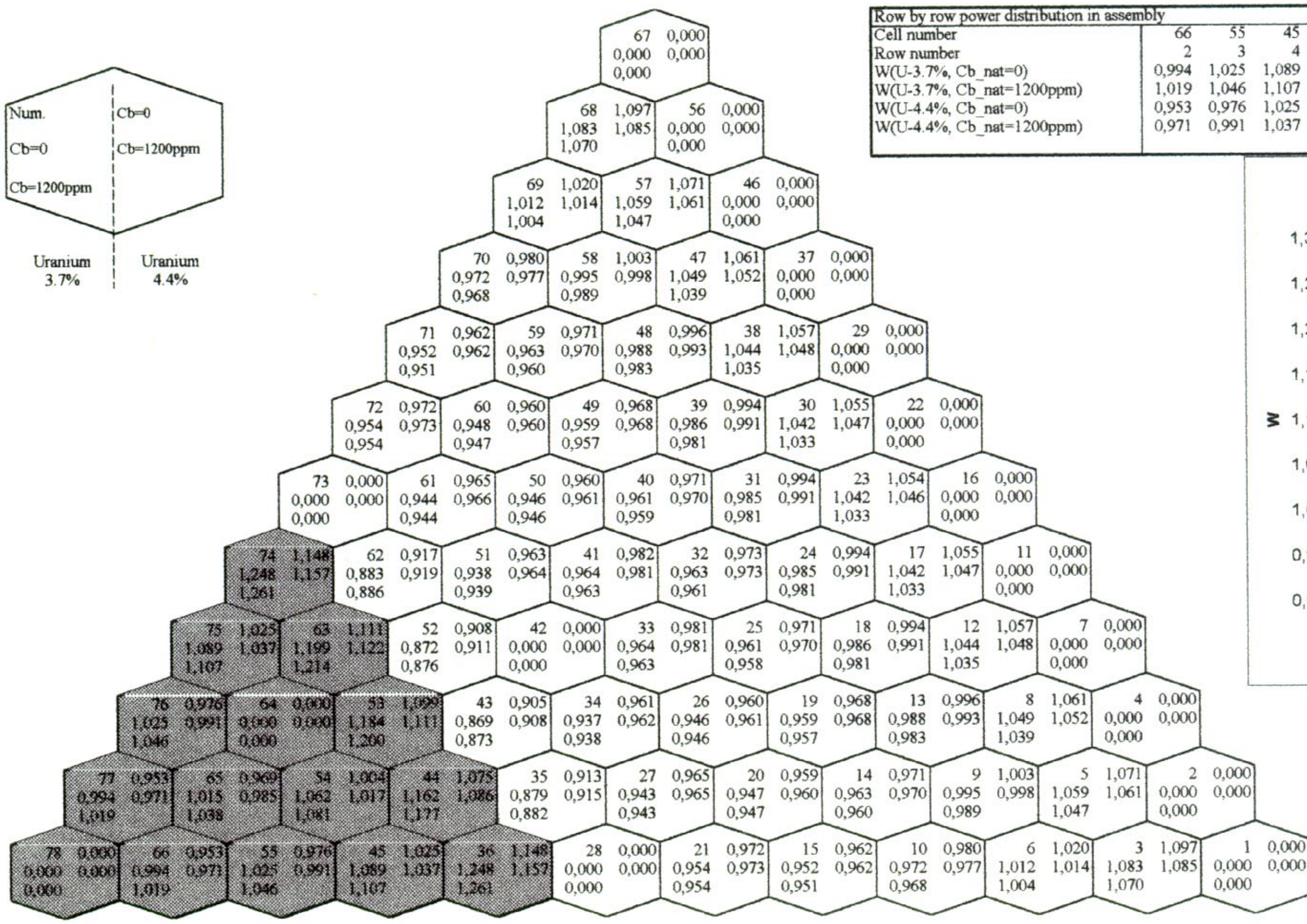


Fig. 2.24. Comparison of Power Inter-assembly Distributions in "Island-1". 3% Plutonium Central Part with 3.7% and 4.4% Uranium Regions



Row by row power distribution in assembly

Cell number	66	55	45	36	28	21	15	10	6	3
Row number	2	3	4	5	6	7	8	9	10	11
W(U-3.7%, Cb_nat=0)	0,994	1,025	1,089	1,248		0,954	0,952	0,972	1,012	1,083
W(U-3.7%, Cb_nat=1200ppm)	1,019	1,046	1,107	1,261		0,954	0,951	0,968	1,004	1,070
W(U-4.4%, Cb_nat=0)	0,953	0,976	1,025	1,148		0,972	0,962	0,980	1,020	1,097
W(U-4.4%, Cb_nat=1200ppm)	0,971	0,991	1,037	1,157		0,973	0,962	0,977	1,014	1,085

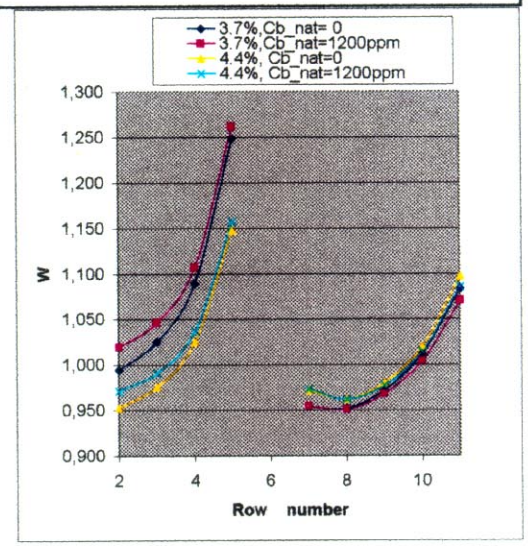
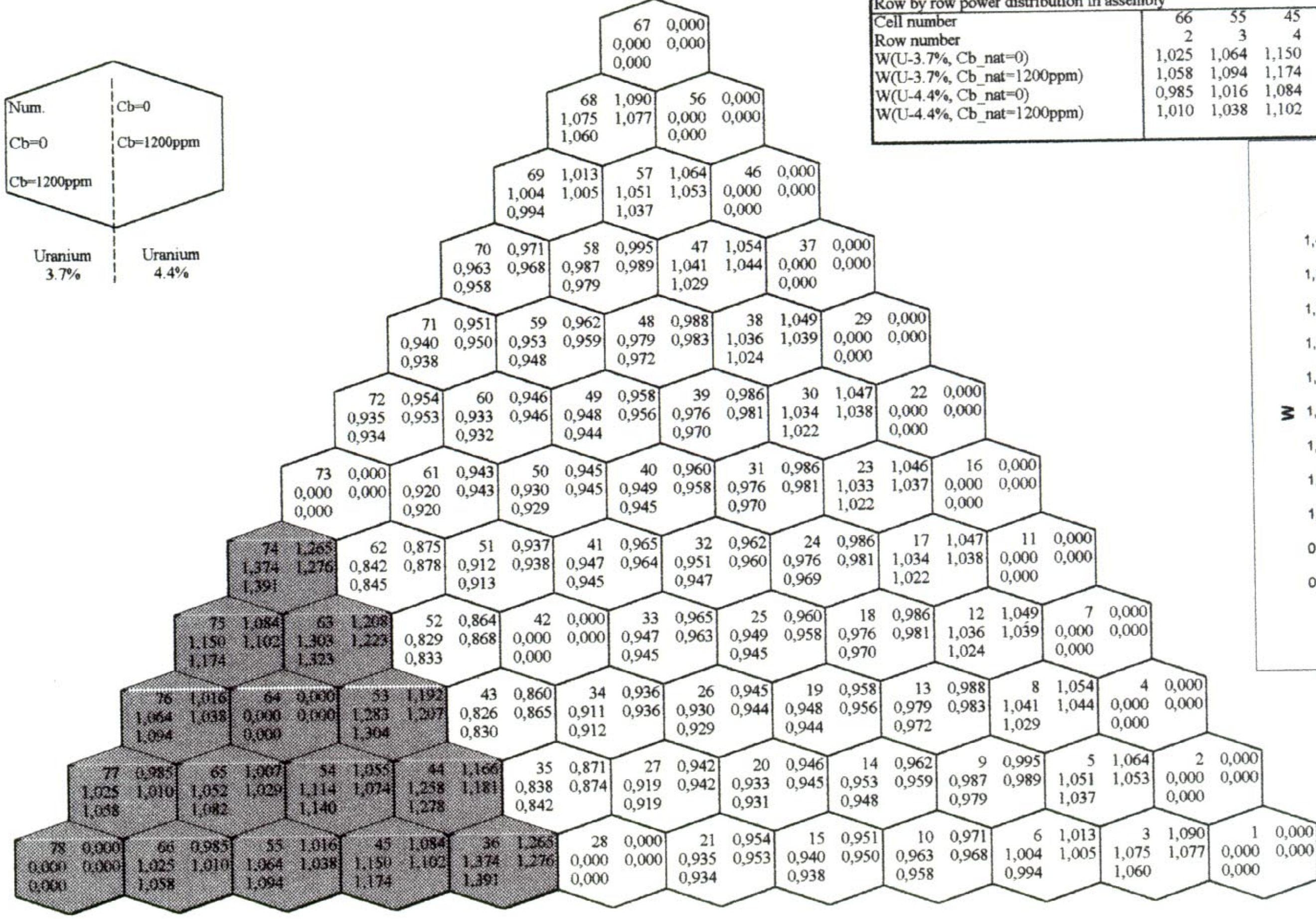


Fig. 2.25. Comparison of Power Inter-assembly Distributions in "Island-1". 4% Plutonium Central Part with 3.7% and 4.4% Uranium Regions



Row by row power distribution in assembly										
Cell number	66	55	45	36	28	21	15	10	6	3
Row number	2	3	4	5	6	7	8	9	10	11
W(U-3.7%, Cb_nat=0)	1,025	1,064	1,150	1,374		0,935	0,940	0,963	1,004	1,075
W(U-3.7%, Cb_nat=1200ppm)	1,058	1,094	1,174	1,391		0,934	0,938	0,958	0,994	1,060
W(U-4.4%, Cb_nat=0)	0,985	1,016	1,084	1,265		0,954	0,951	0,971	1,013	1,090
W(U-4.4%, Cb_nat=1200ppm)	1,010	1,038	1,102	1,276		0,953	0,950	0,968	1,005	1,077

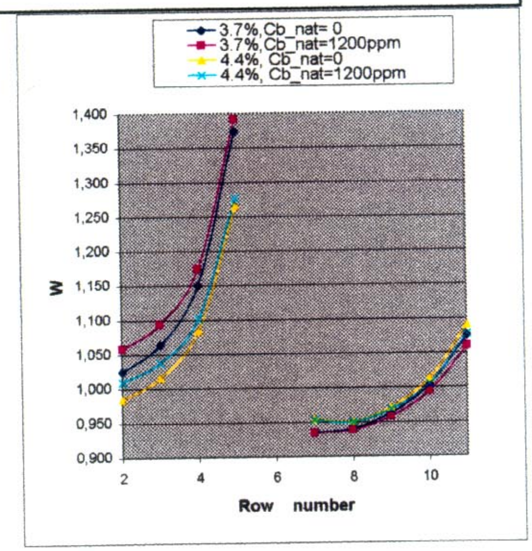
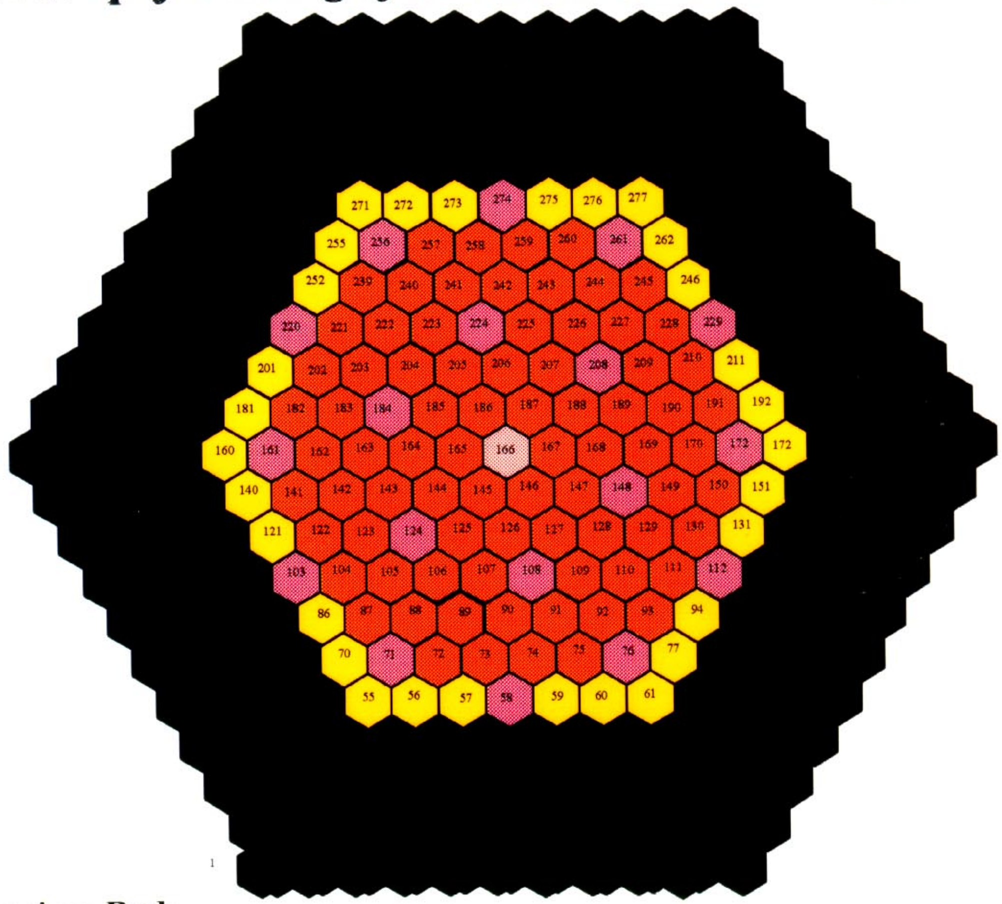


Figure 2.26 . Simplified Design for "Increased Island-2" Type MOX LTA



Enriched Uranium Rods

High Plutonium-Content MOX Rods

Intermediate Plutonium-Content MOX Rods

Central tube

Guide Tube

Fig. 2.27 Kk against "Island" periphery enrichment for different "Island" size. "Island central enrichment - 4.0%. Uranium enrichment - 3.7%.

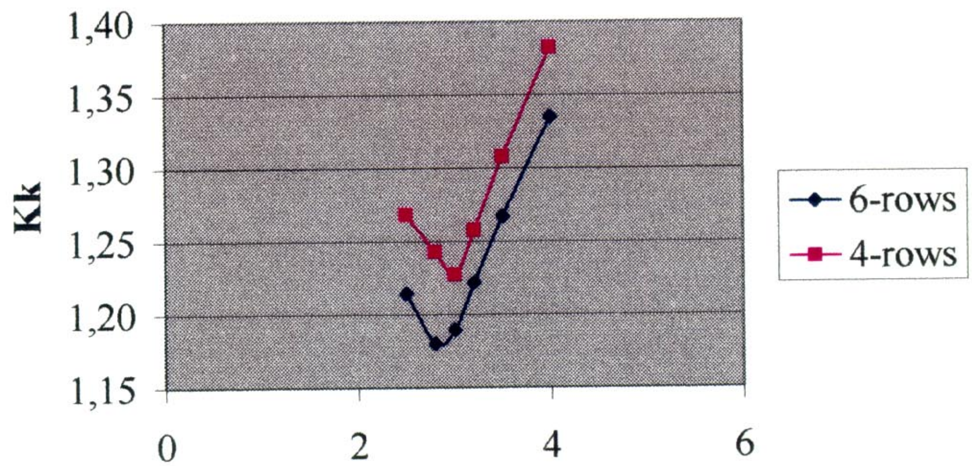
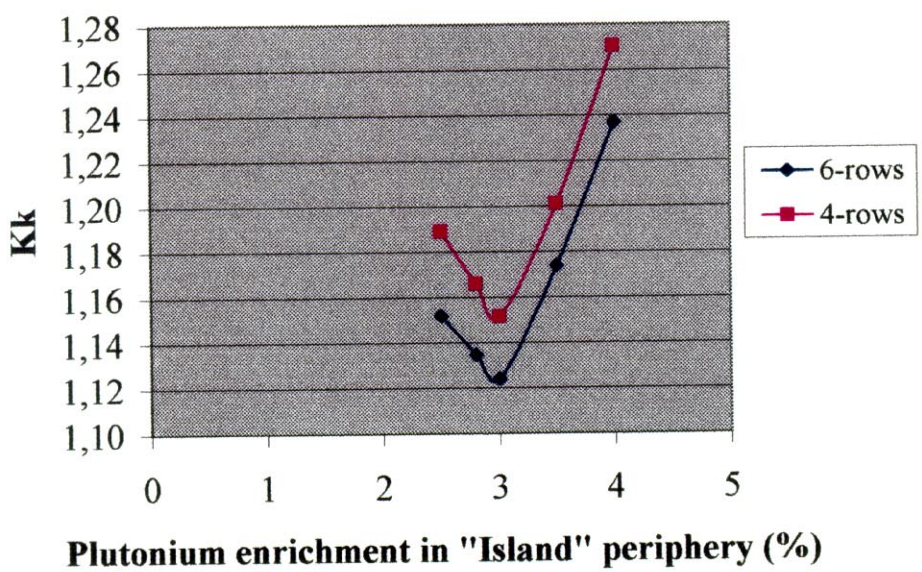


Fig. 2.28 Kk against "Island" periphery enrichment for different "Island" size. "Island central enrichment - 4.0%. Uranium enrichment - 4.4%.



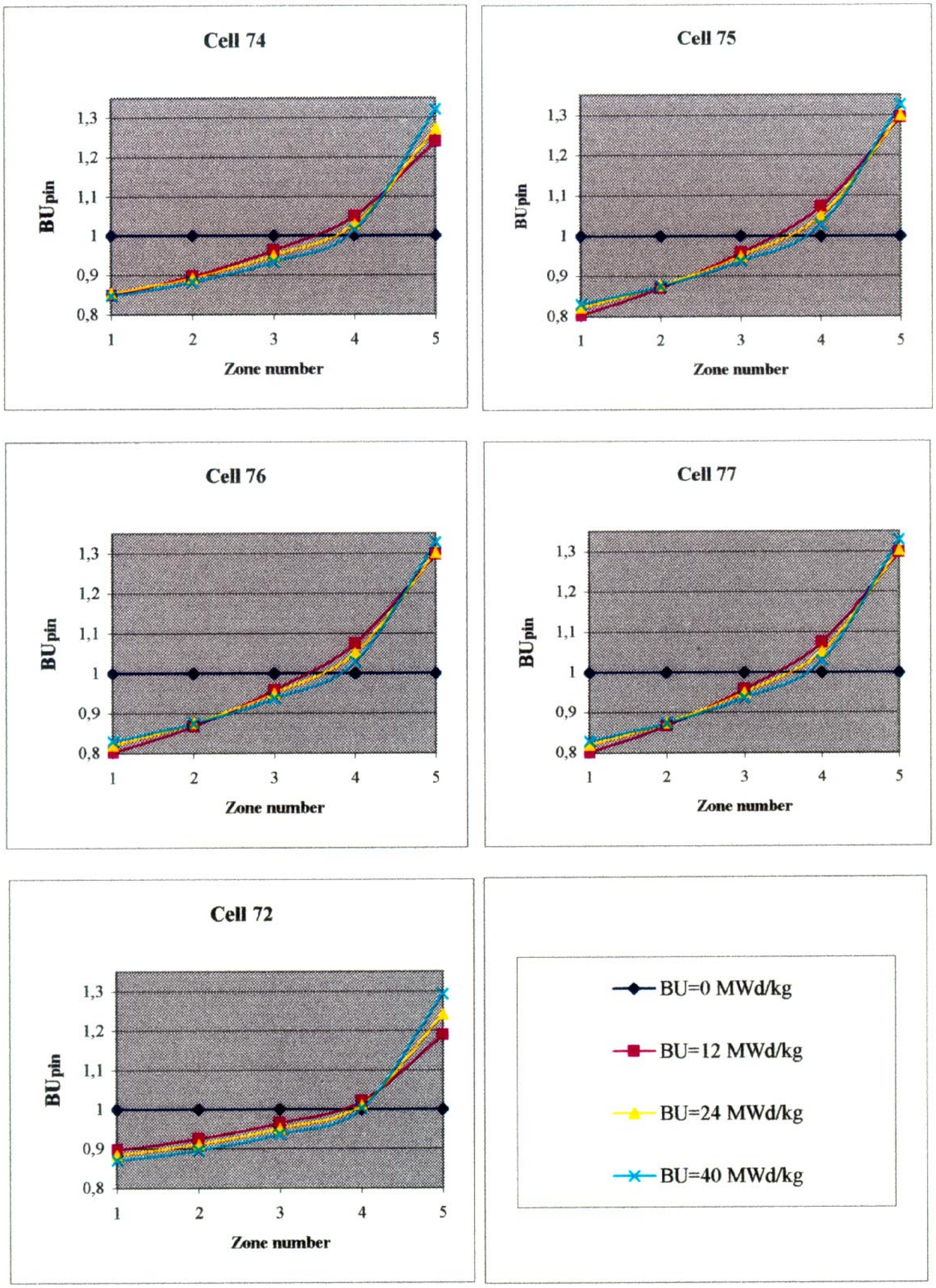


Fig. 2.29 Inter-pin relative burnup distribution

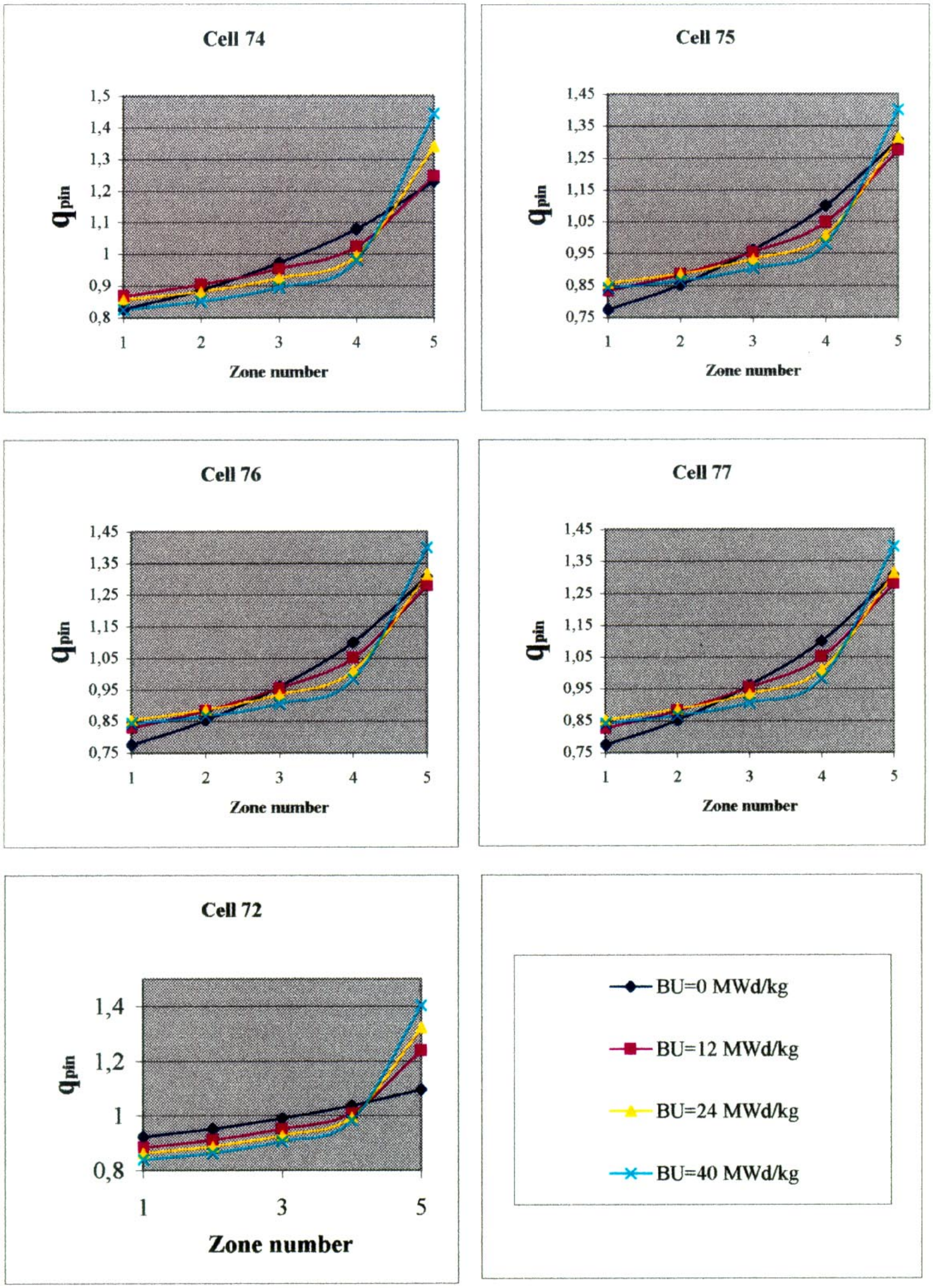


Fig. 2.30 Inter-pin relative power distribution

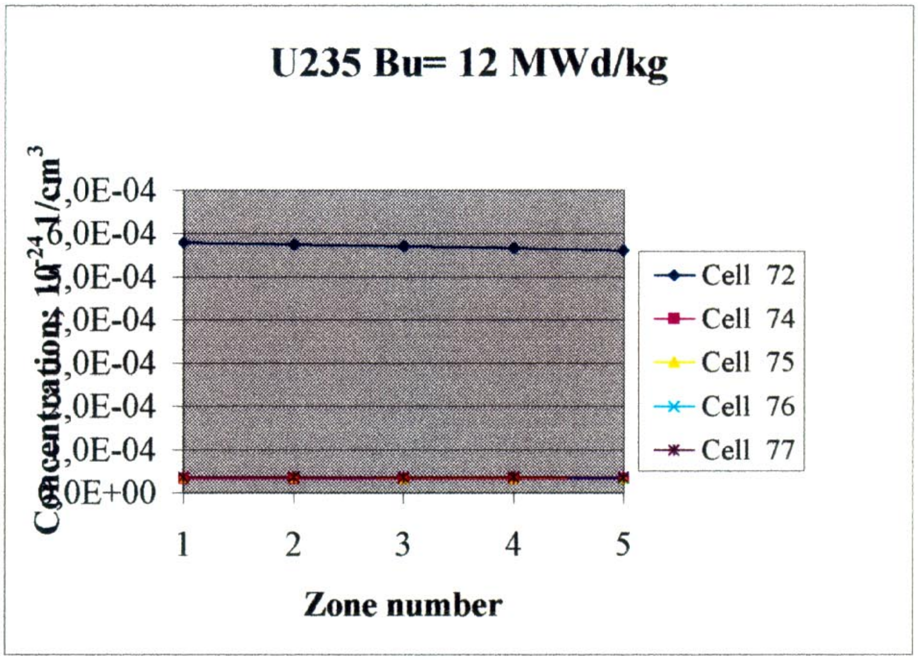


Fig. 2.31. Inter-pin isotopic distribution

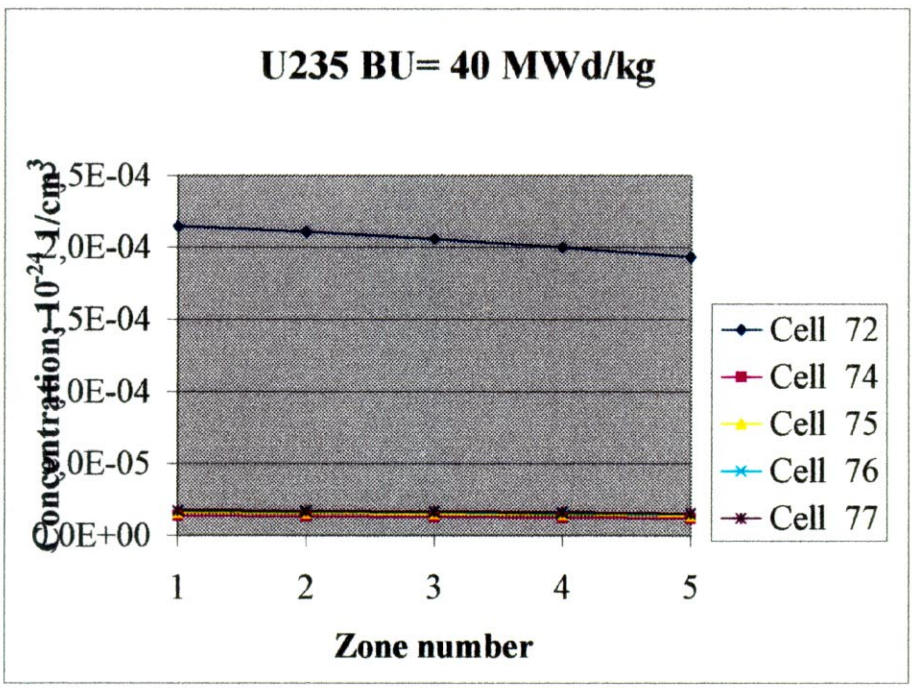


Fig. 2.32. Inter-pin isotopic distribution

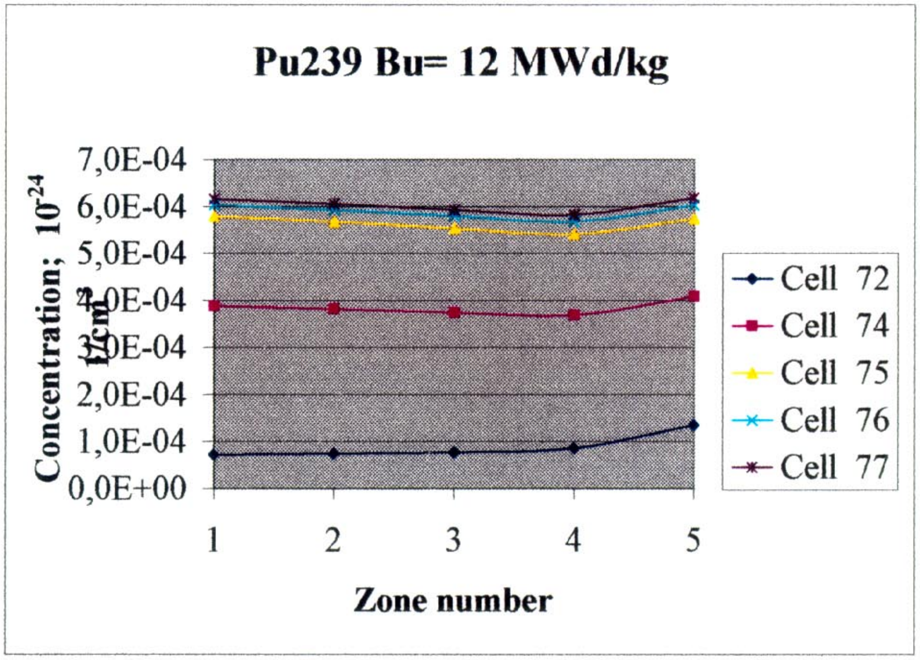


Fig. 2.33 Inter-pin isotopic distribution

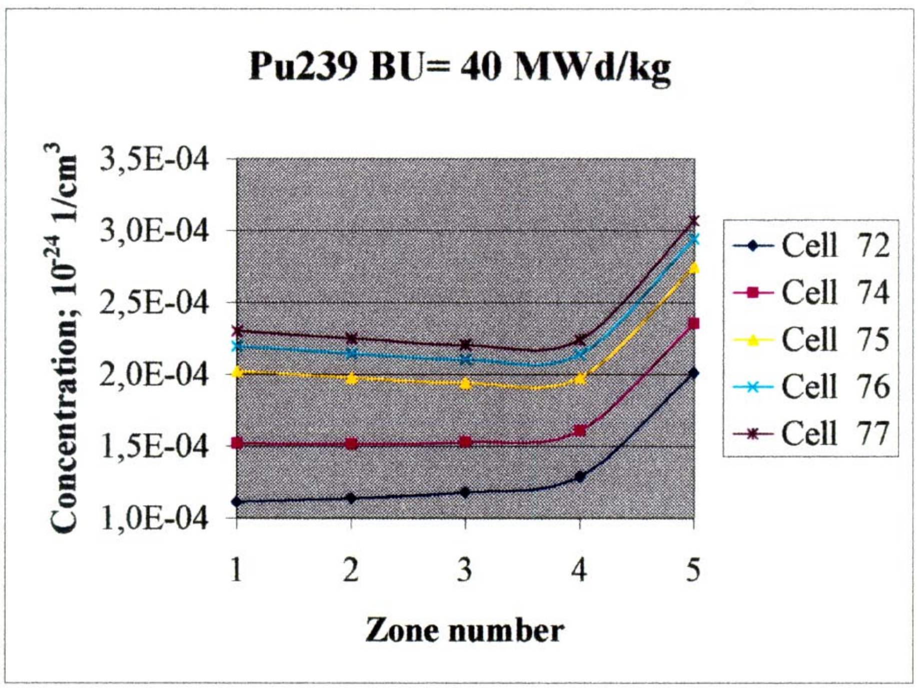


Fig. 2.34 Inter-pin isotopic distribution

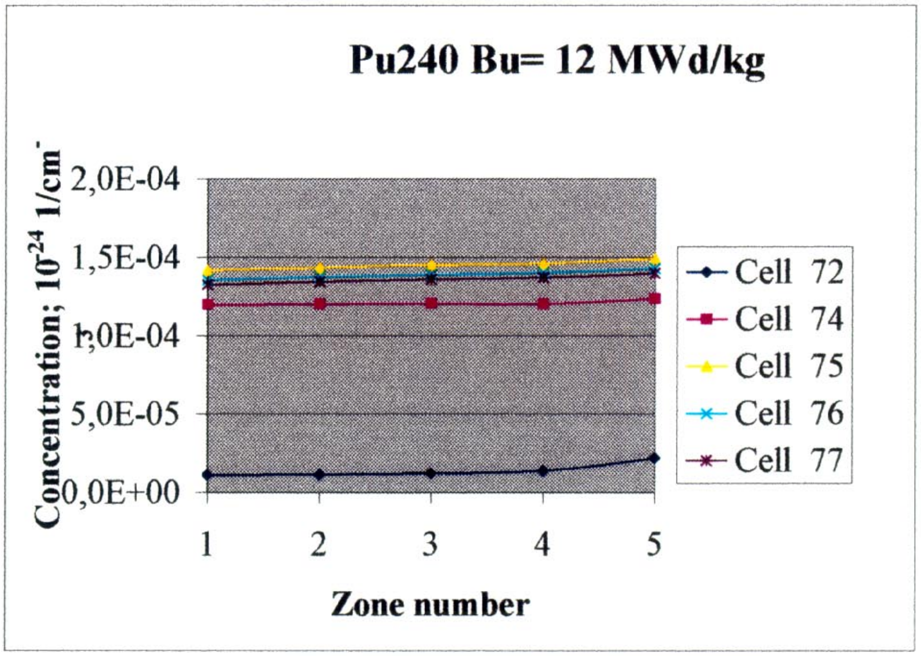


Fig. 2.35 Inter-pin isotopic distribution

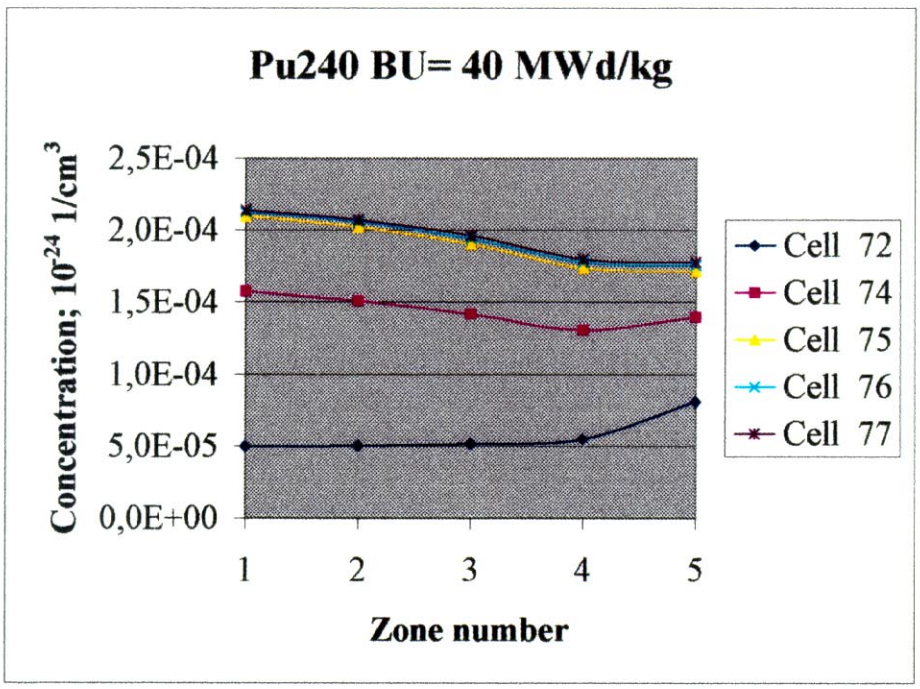


Fig. 2.36 Inter-pin isotopic distribution

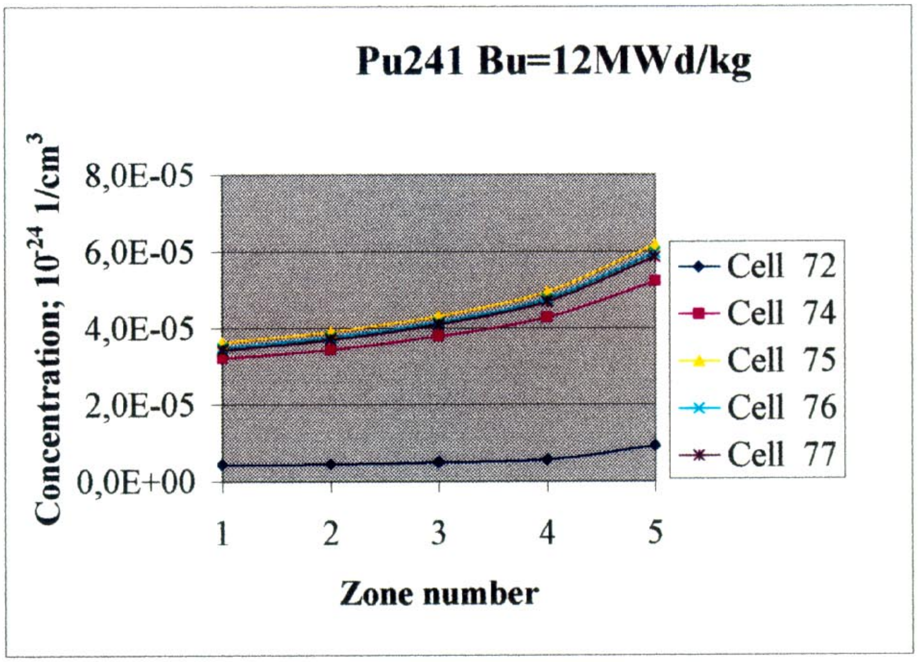


Fig. 2.37 Inter-pin isotopic distribution

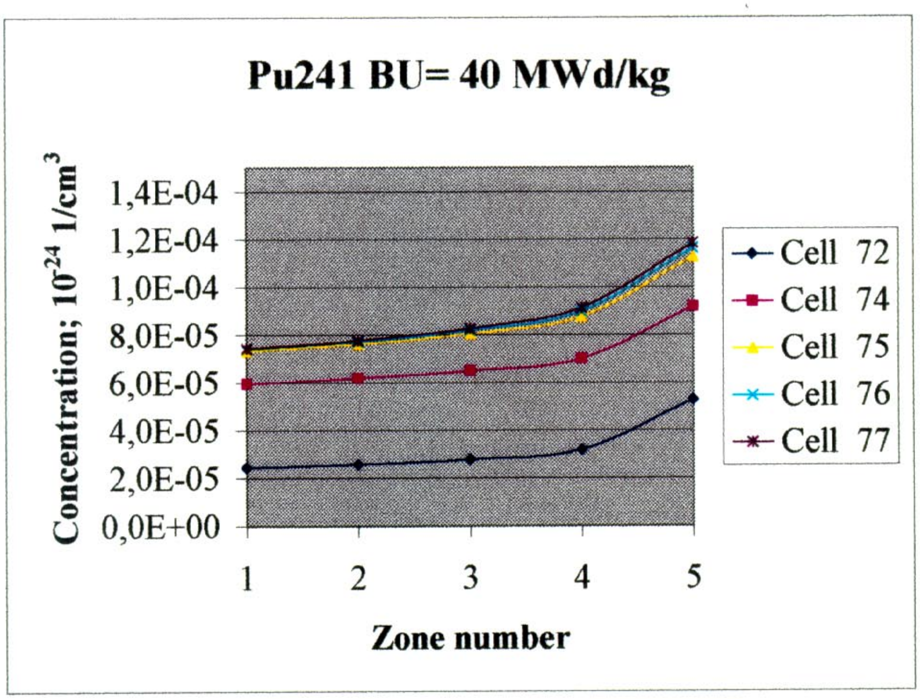


Fig. 2.38 Inter-pin isotopic distribution

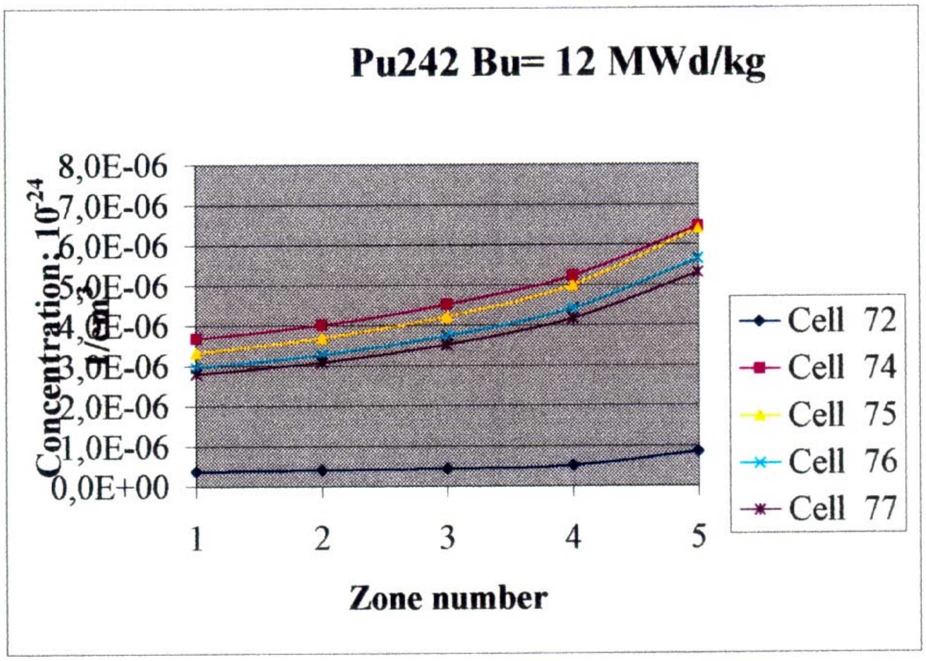


Fig. 2.39 Inter-pin isotopic distribution

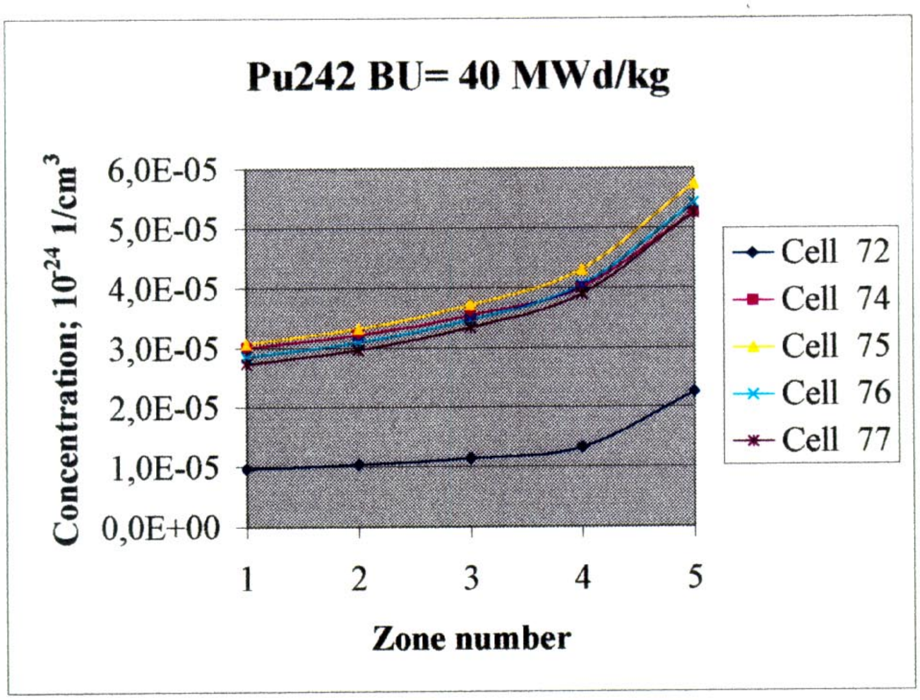


Fig. 2.40 Inter-pin isotopic distribution

H:\SERV_AM1\DATA\pU38_pu38_so

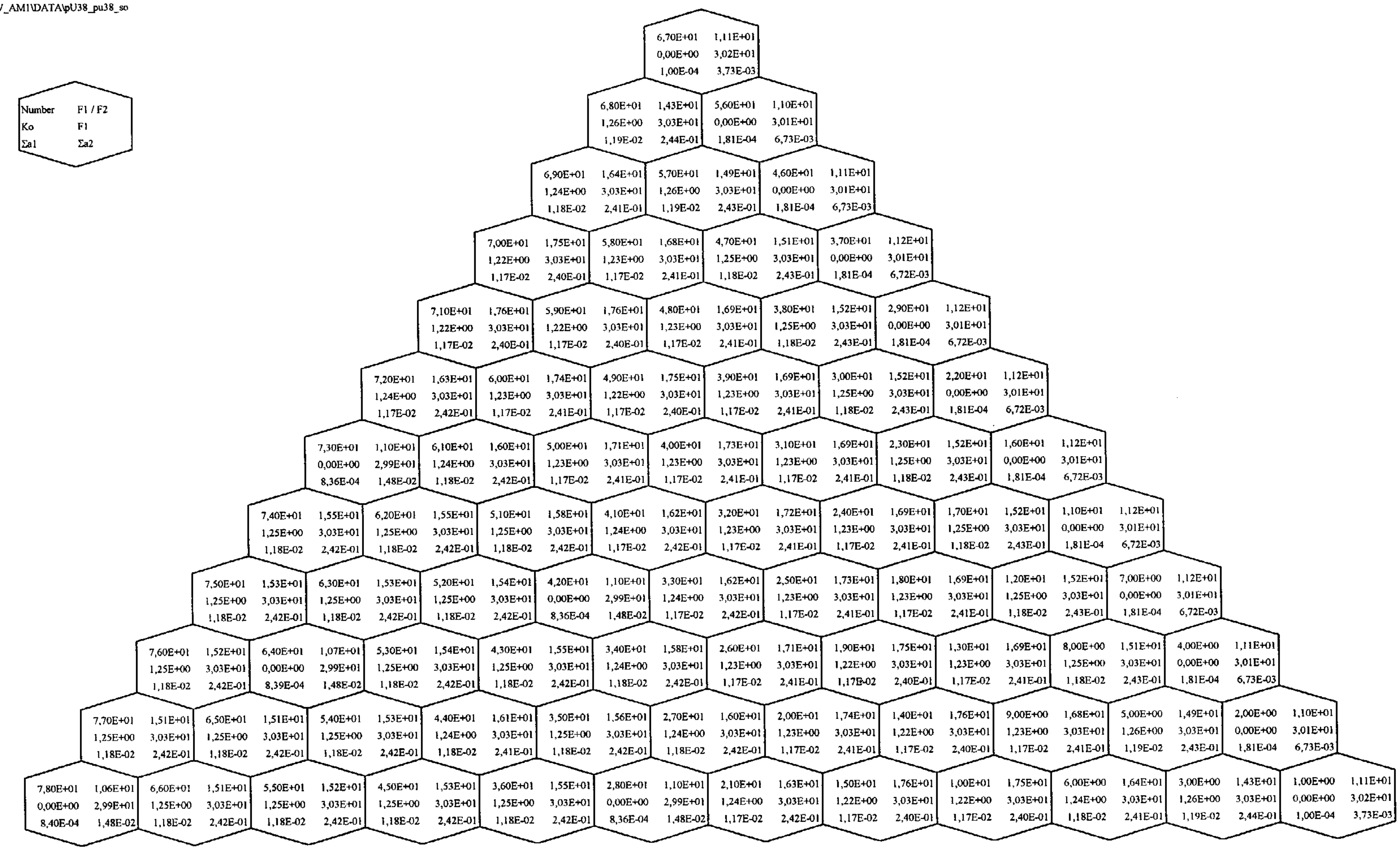


Fig. 2.41 Spectrum parameters distribution in MOX assembly (Pu 3.8. Sector 60°)

H:\SERV_AMI\DATA\pU38_38_u37_so

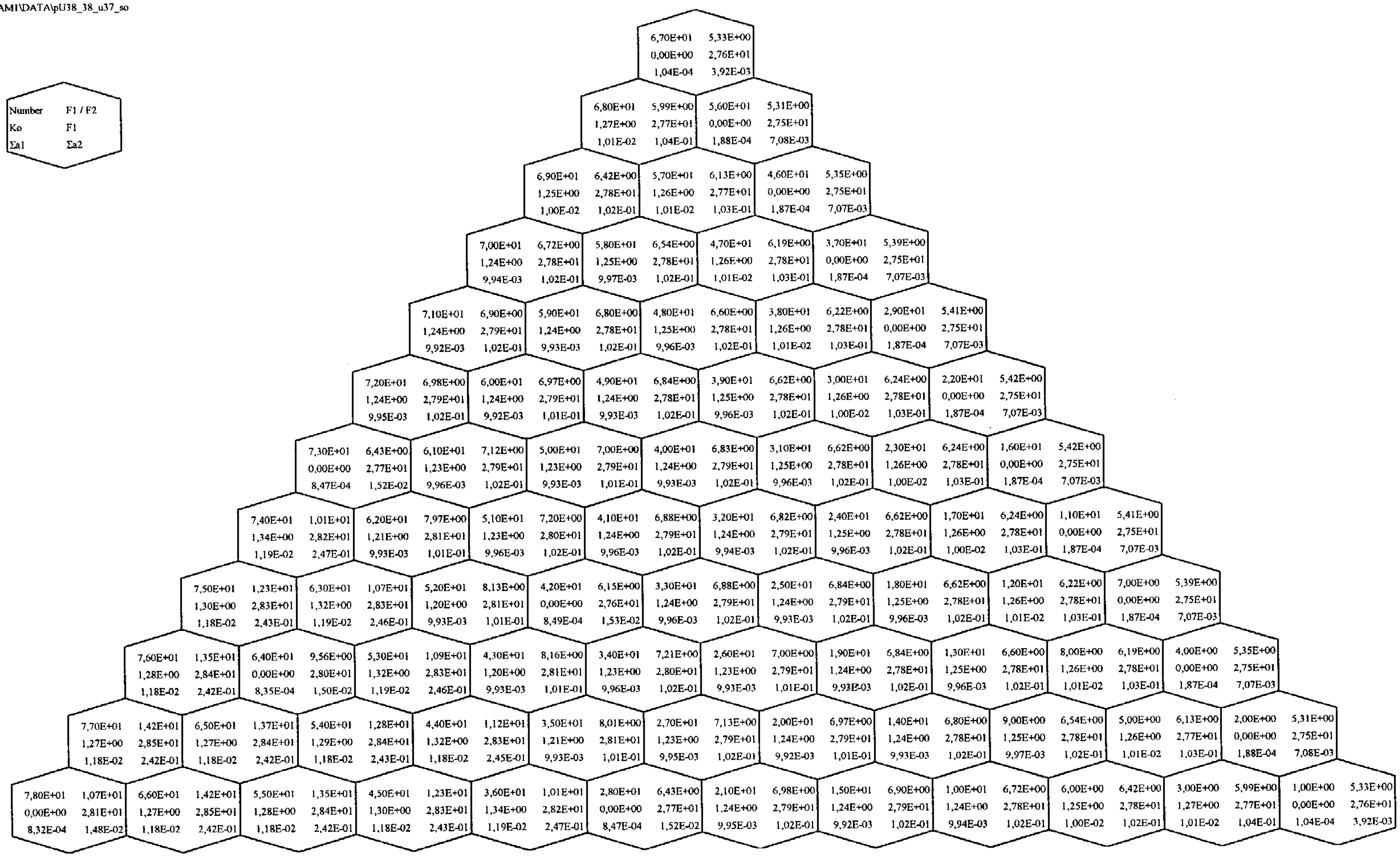


Fig. 2.42 Spectrum parameters distribution in "Island" type MOX assembly (Pu 3.8_3.8_U 3.7. Sector 60°)

H:\SERV_AMI\DATA\pU38_28_u37_so

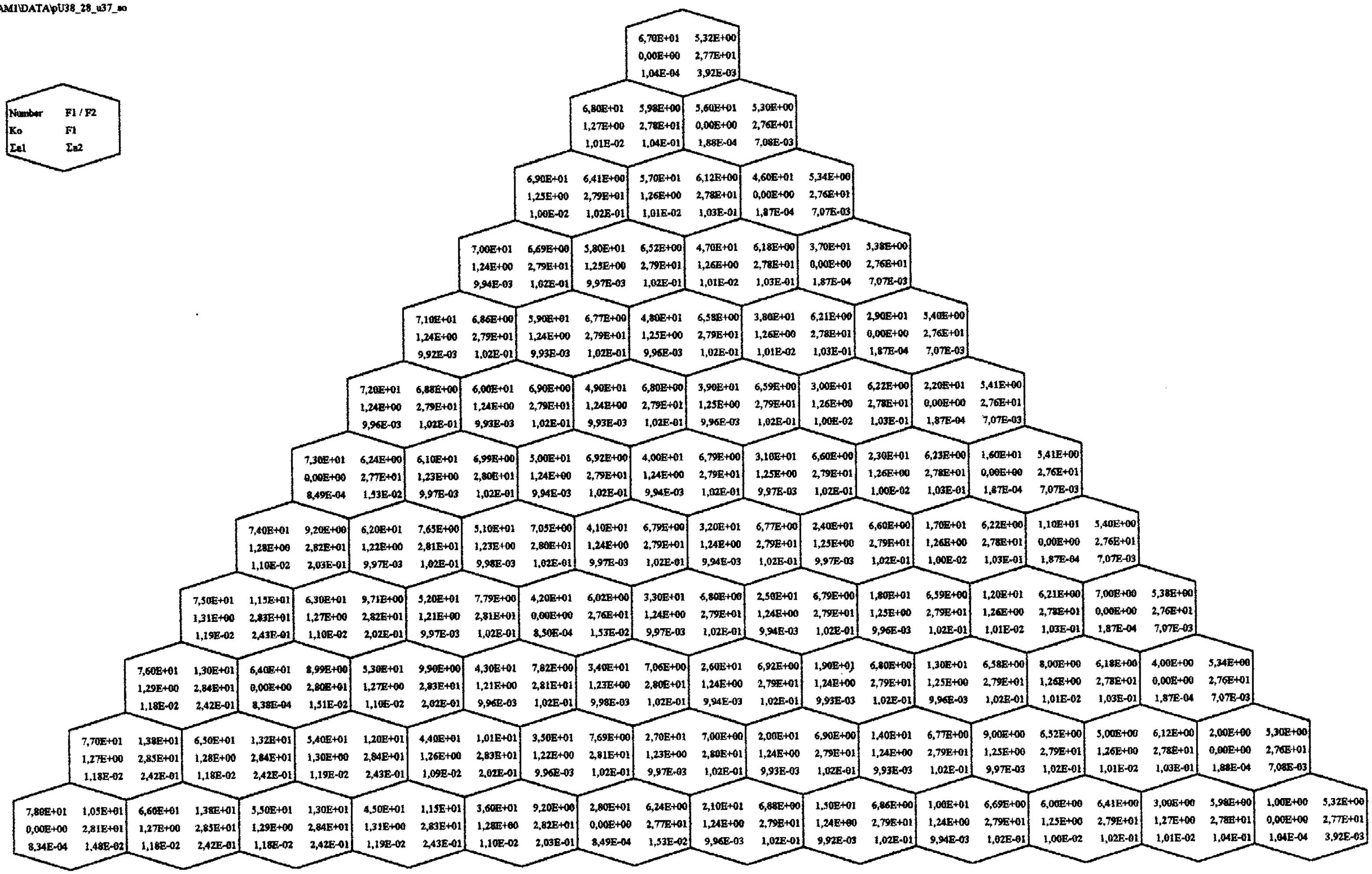


Fig. 2.43 Spectrum parameters distribution in "Island" type MOX assembly (Pu 3.8_2.8_U 3.7. Sector 60o)

pu38_28_u37o
Current Burnup 0 MWtd/kg
Power Distribution

										68
										1,073
									69	57
									1,005	1,05
								70	58	47
								0,968	0,989	1,041
							71	59	48	38
							0,949	0,959	0,983	1,037
						72	60	49	39	30
						0,951	0,944	0,955	0,98	1,035
					73	61	50	40	31	23
					0	0,94	0,943	0,957	0,98	1,034
				74	62	51	41	32	24	17
				1,19	0,88	0,935	0,96	0,959	0,98	1,035
			75	63	52	42	33	25	18	12
			1,219	1,137	0,869	0	0,96	0,957	0,98	1,037
		76	64	53	43	34	26	19	13	8
		1,108	0	1,121	0,865	0,933	0,943	0,955	0,983	1,041
	77	65	54	44	35	27	20	14	9	5
	1,058	1,092	1,175	1,1	0,876	0,939	0,944	0,958	0,989	1,05
78	66	55	45	36	28	21	15	10	6	3
0	1,058	1,108	1,219	1,19	0	0,951	0,949	0,968	1,005	1,073

pu38_28_u37o
Current Burnup 12 MWtd/kg
Power Distribution

										68
										1,056
									69	57
									1,005	1,039
								70	58	47
								0,975	0,993	1,032
							71	59	48	38
							0,962	0,968	0,987	1,028
						72	60	49	39	30
						0,973	0,962	0,967	0,986	1,027
					73	61	50	40	31	23
					0	0,972	0,964	0,969	0,986	1,026
				74	62	51	41	32	24	17
				1,018	0,95	0,972	0,979	0,971	0,986	1,027
			75	63	52	42	33	25	18	12
			1,155	1,008	0,945	0	0,979	0,969	0,986	1,028
		76	64	53	43	34	26	19	13	8
		1,106	0	1,002	0,942	0,971	0,963	0,967	0,987	1,032
	77	65	54	44	35	27	20	14	9	5
	1,077	1,097	1,136	0,991	0,947	0,971	0,962	0,968	0,992	1,039
78	66	55	45	36	28	21	15	10	6	3
0	1,077	1,106	1,155	1,018	0	0,973	0,962	0,975	1,005	1,056

Fig. 2.44 Power distribution evolution in "Island" type MOX assembly (Pu3.8_2.8_U3.7 Sector 60°)

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. 2.45 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

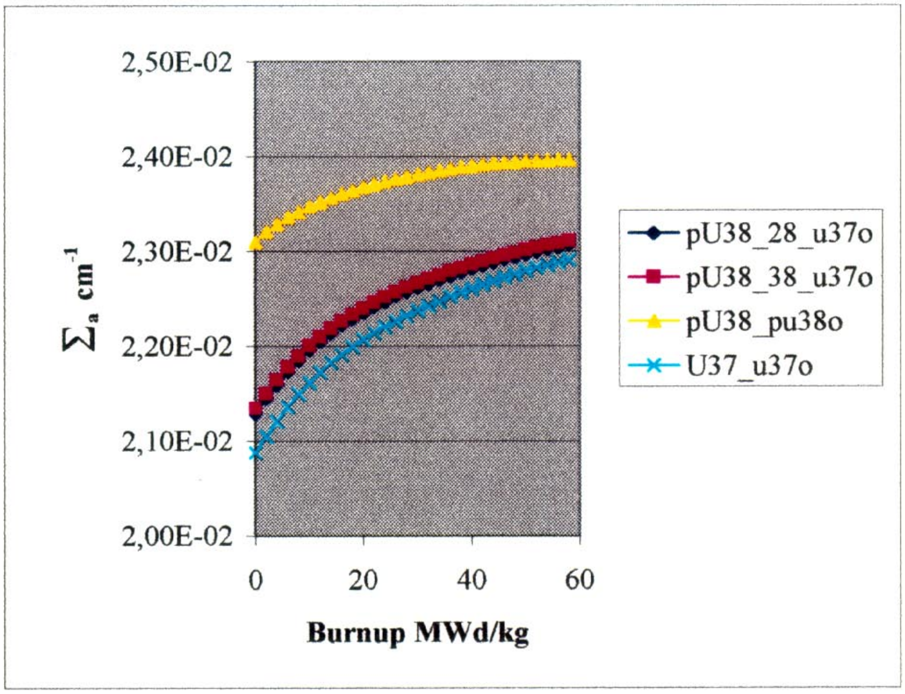
pu38_28_u37o
Current Burnup 24 MWtd/kg
Burnup Distribution (MWtd/kg)

										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. 2.46 Burnup distribution evolution in "Island" type MOX assembly (Pu3.8 2.8 U3.7 Sector 60°)



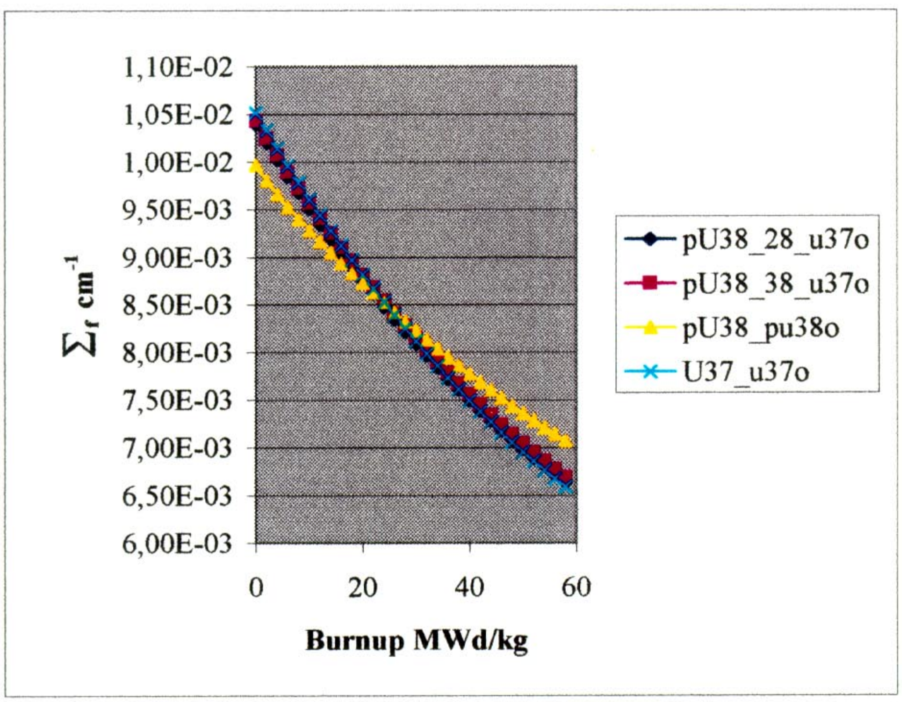
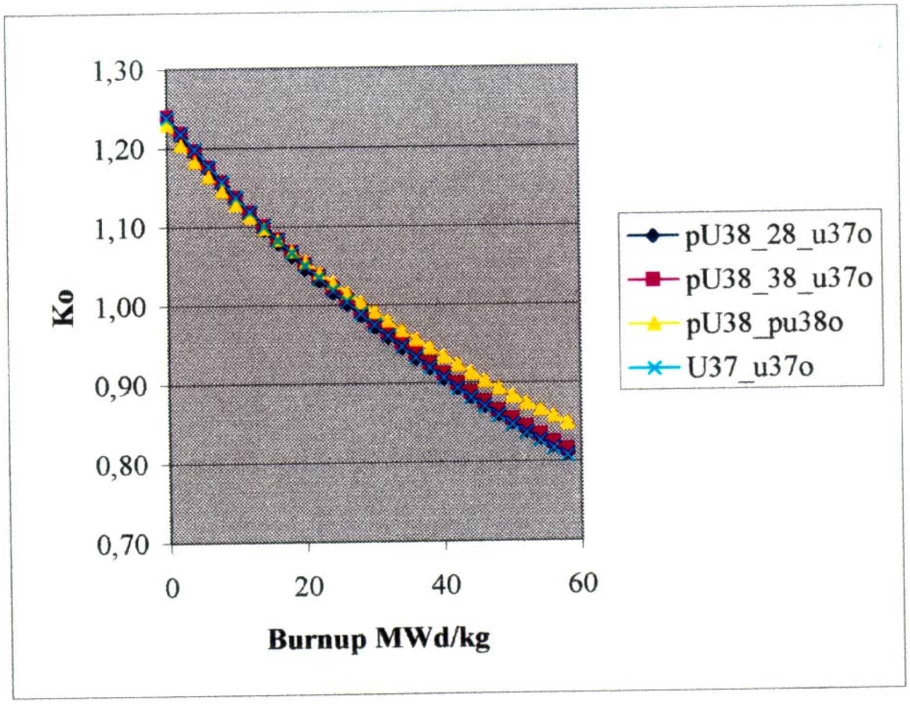


Fig. 2.47 Assembly parameters evolution for different enrichment compositions



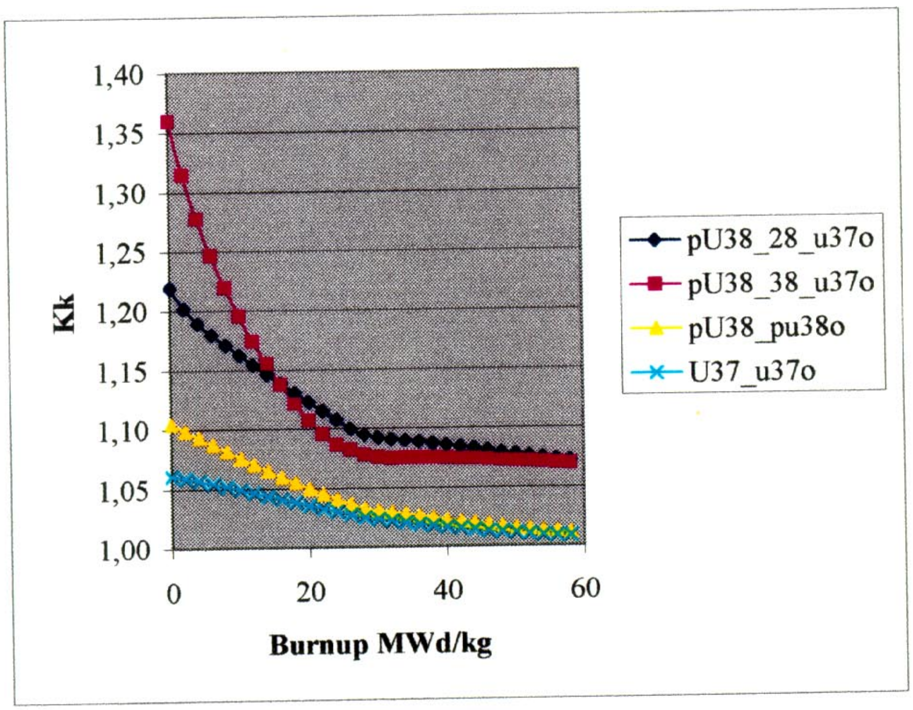


Fig. 2.48 Assembly parameters evolution for different enrichment compositions

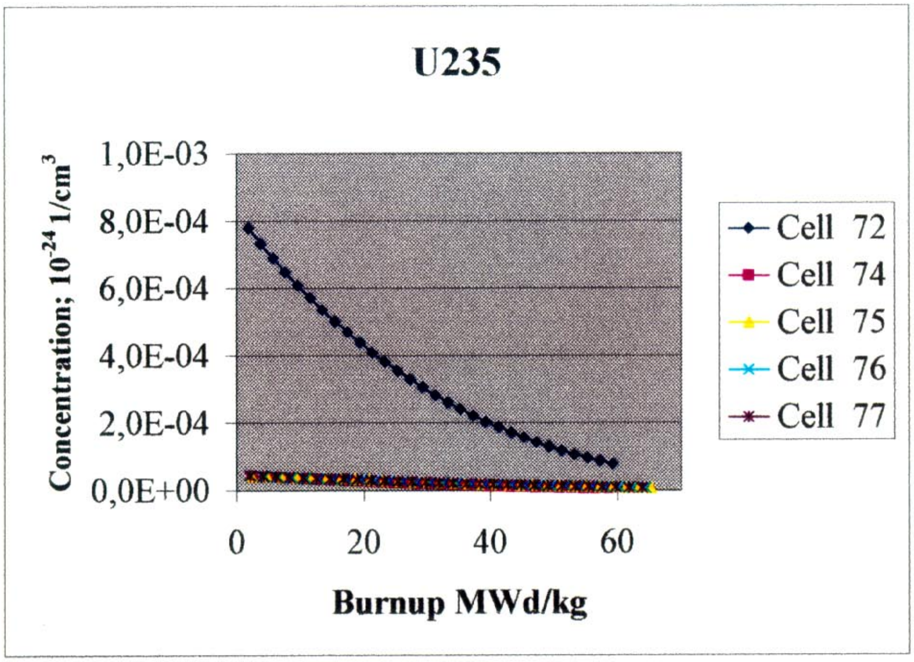


Fig. 2.49 Evolution of pin isotopic content

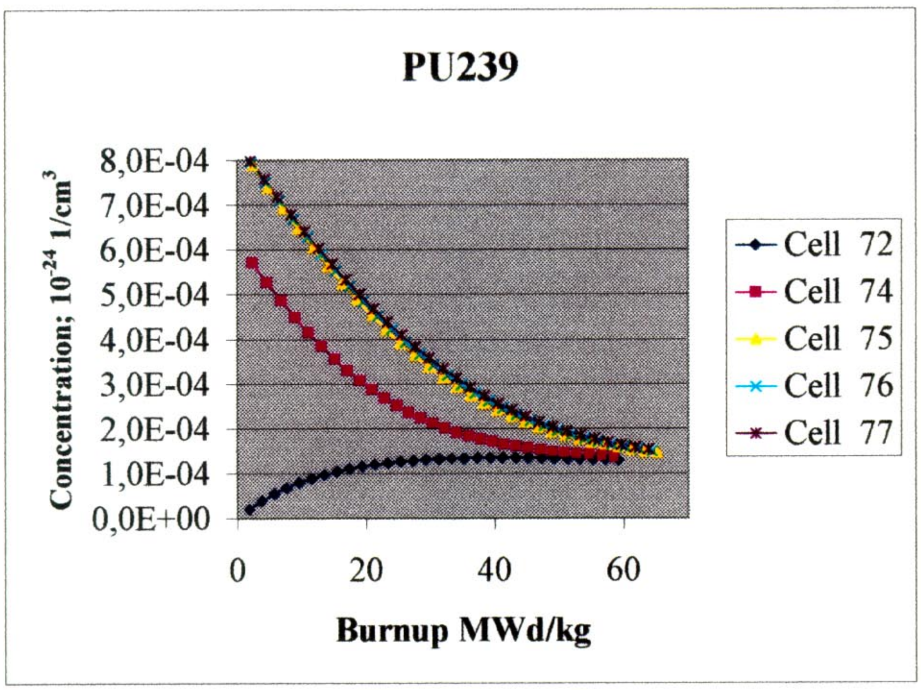


Fig. 2.50 Evolution of pin isotopic content

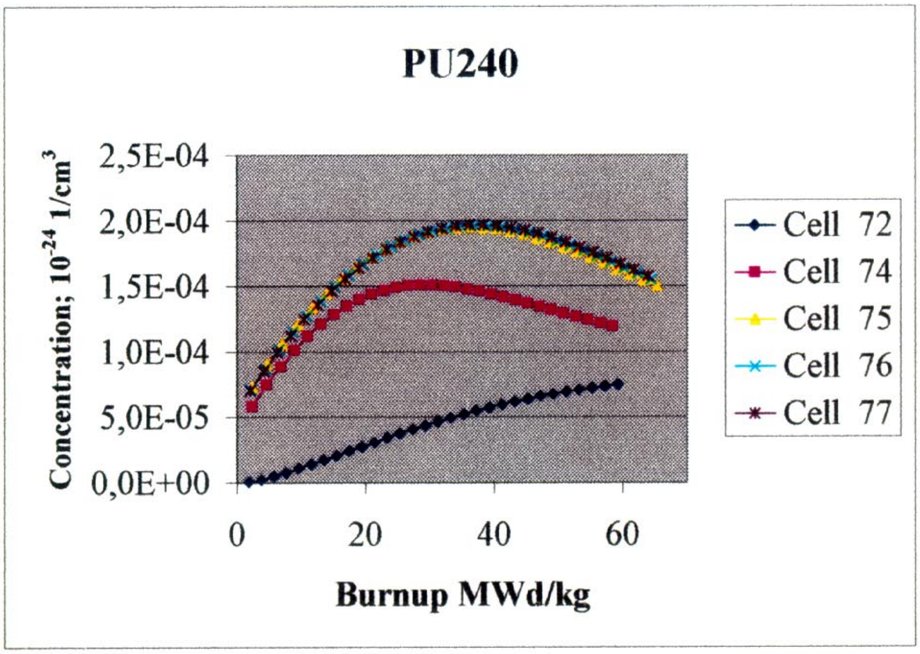


Fig. 2.51 Evolution of pin isotopic content

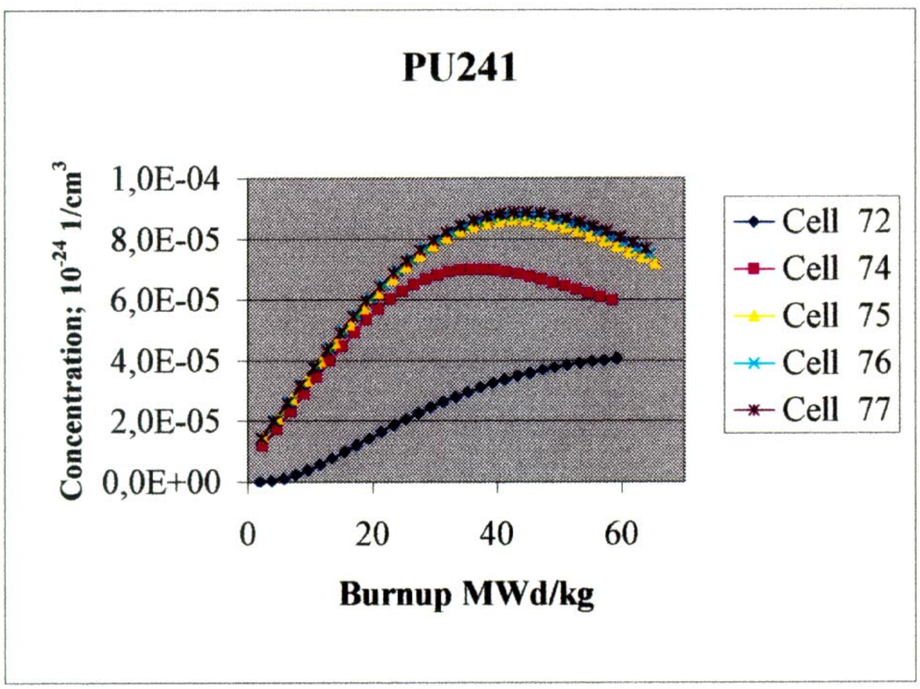


Fig. 2.52 Evolution of pin isotopic content

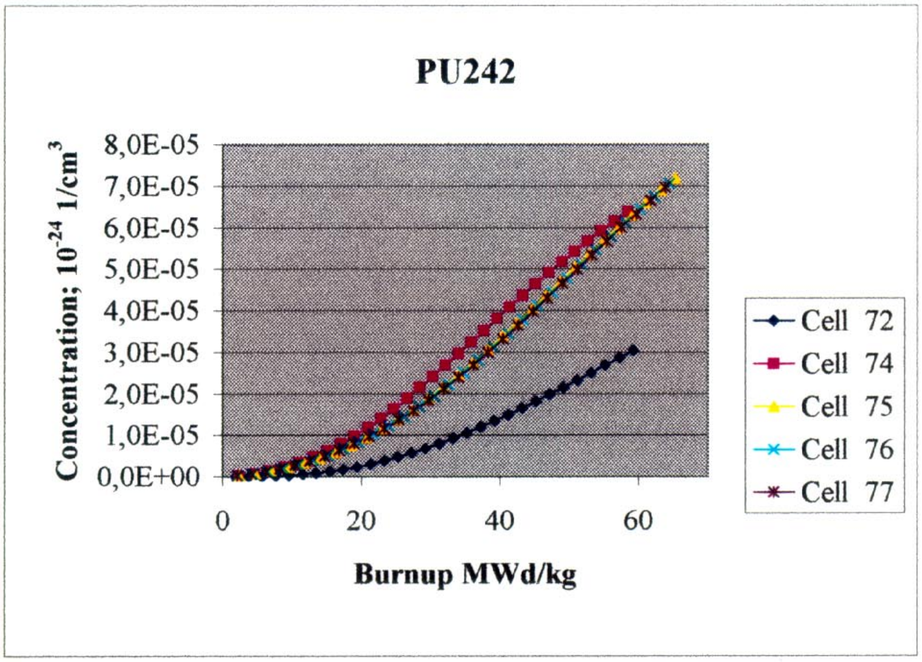


Fig. 2.53 Evolution of pin isotopic content

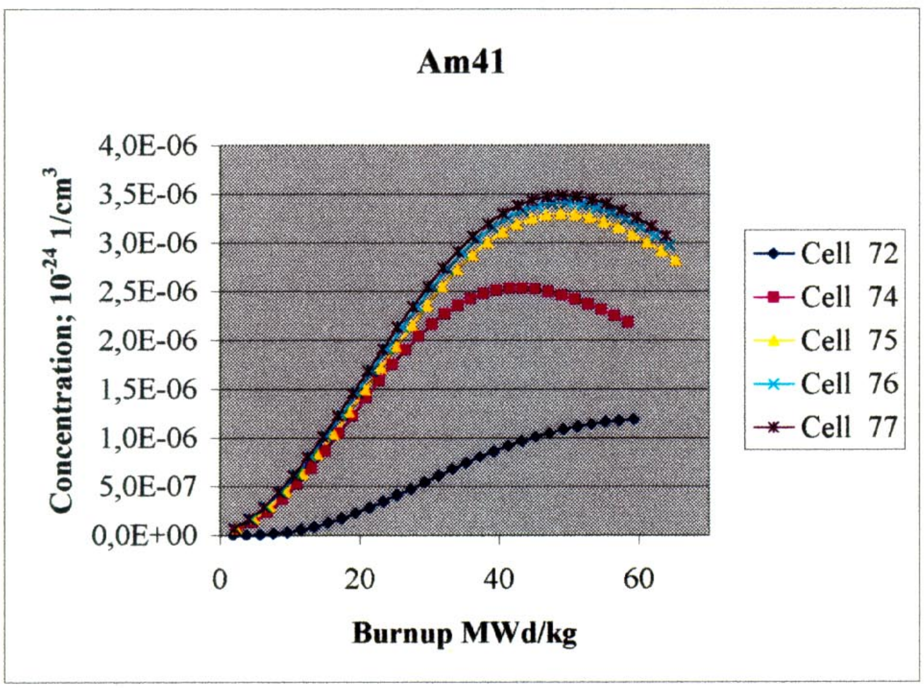


Fig. 2.54 Evolution of pin isotopic content

Fig.3.1. Assembly-by-Assembly Burnup, Power and Temperature Drops Distributions. Equilibrium Cycle for Uranium Reference Core with Boron BPRs. Core 60° Sector

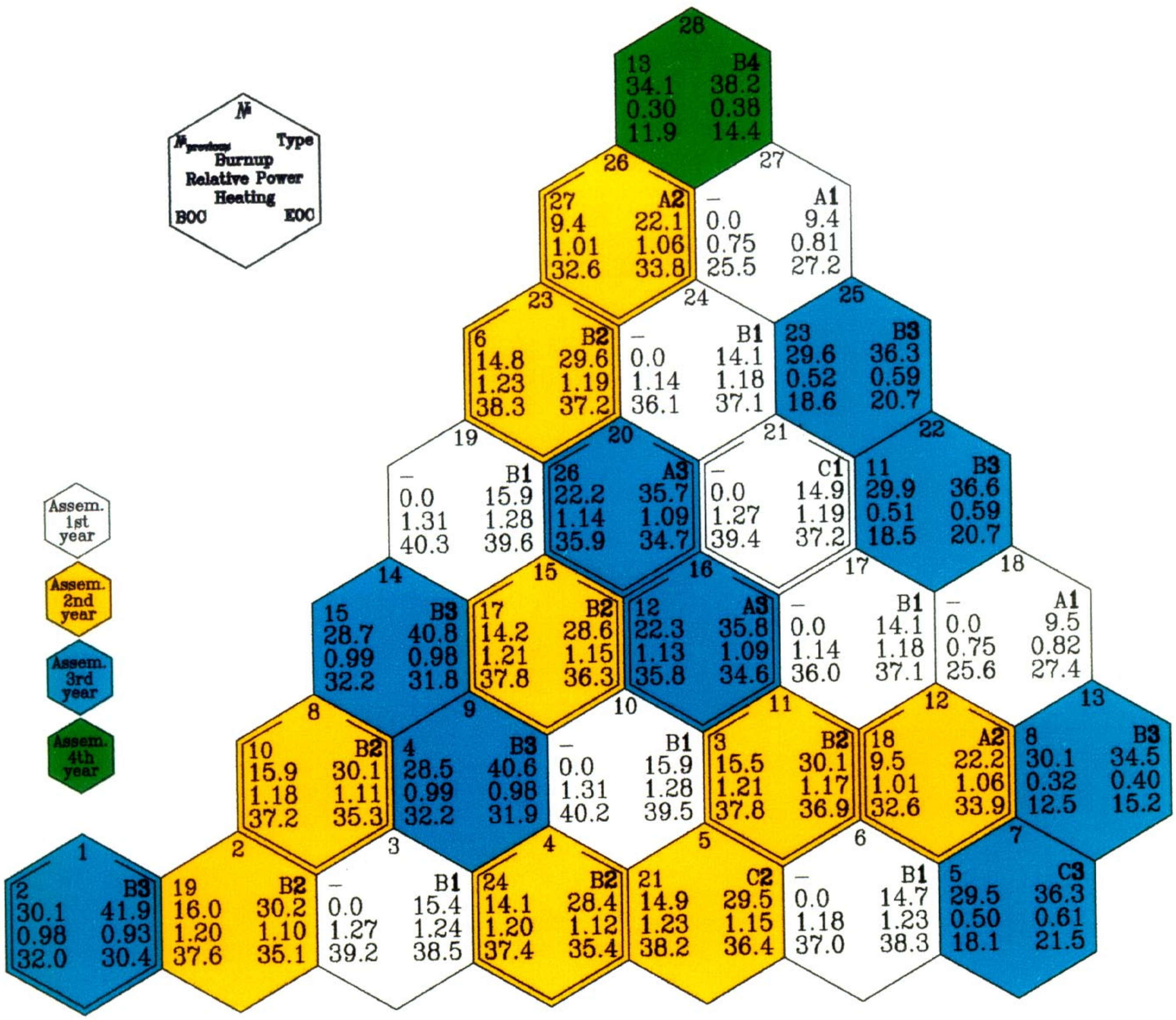


Fig.3.2. Assembly-by-Assembly Maximum Linear Pin Power Distribution in BOC. Equilibrium Cycle for Uranium Reference Core with Boron BPRs. Core 60° Sector

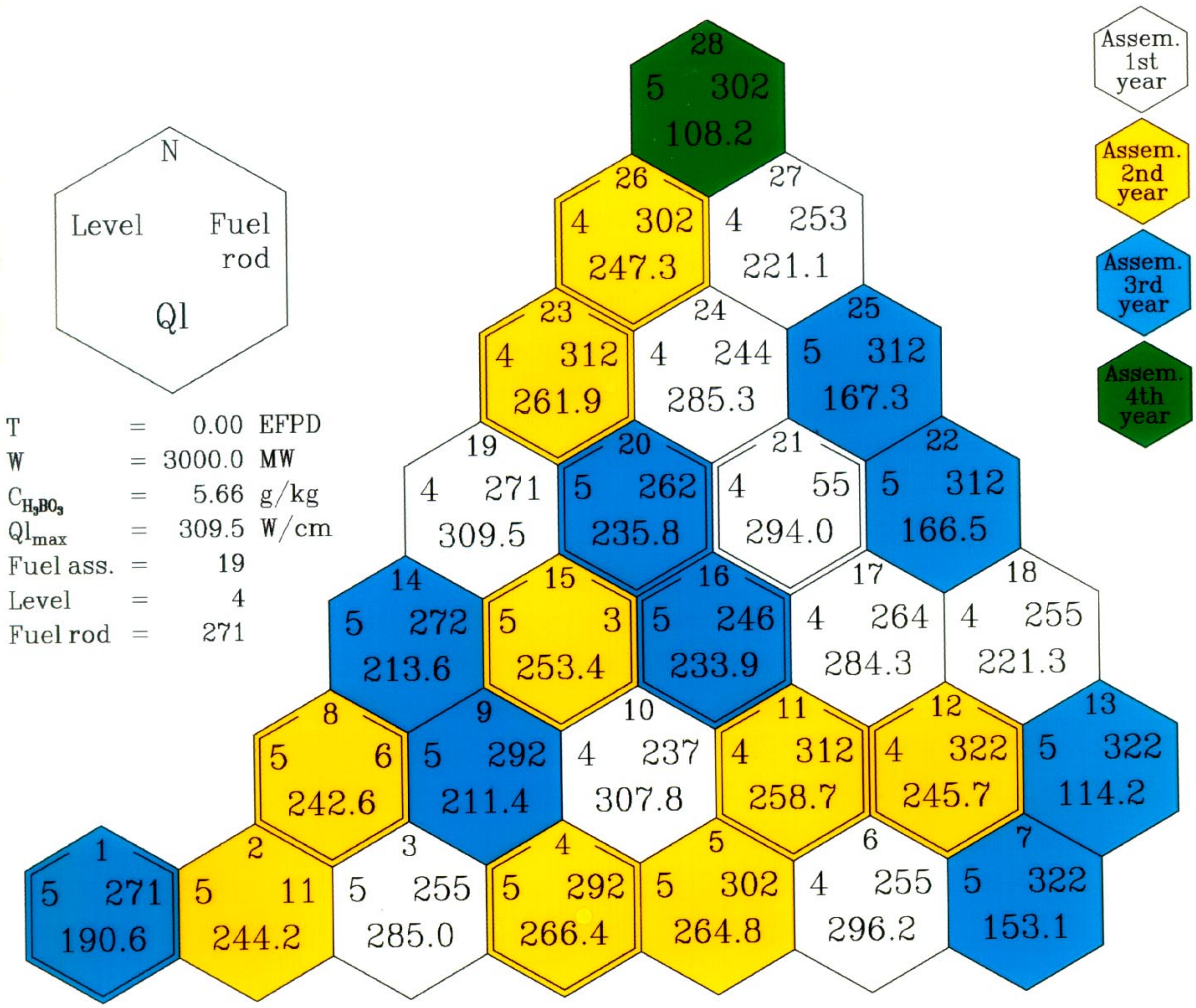


Fig.3.3. Assembly-by-Assembly Maximum Linear Pin Power Distribution in EOC. Equilibrium Cycle for Uranium Reference Core with Boron BPRs.Core 60° Sector

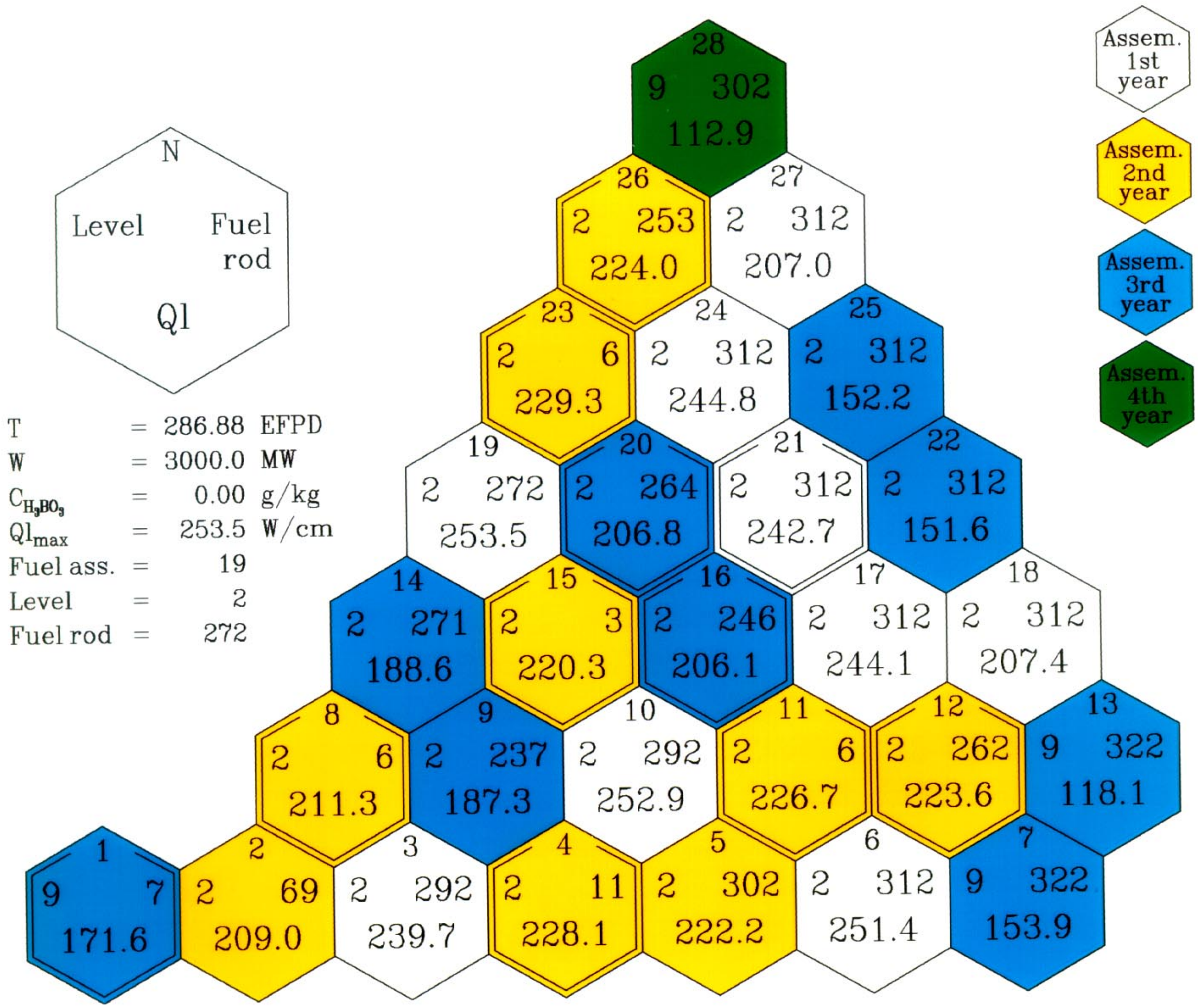
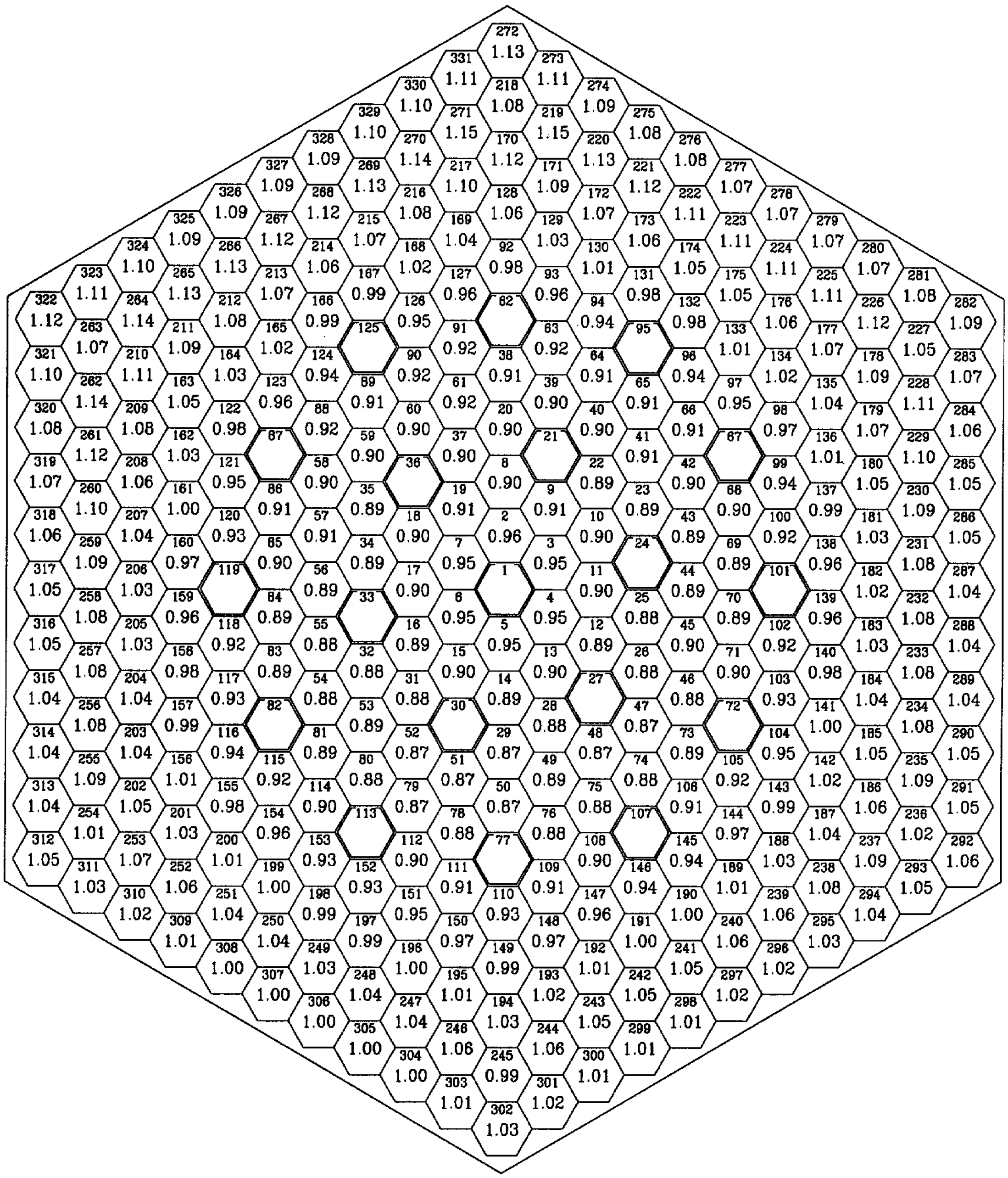
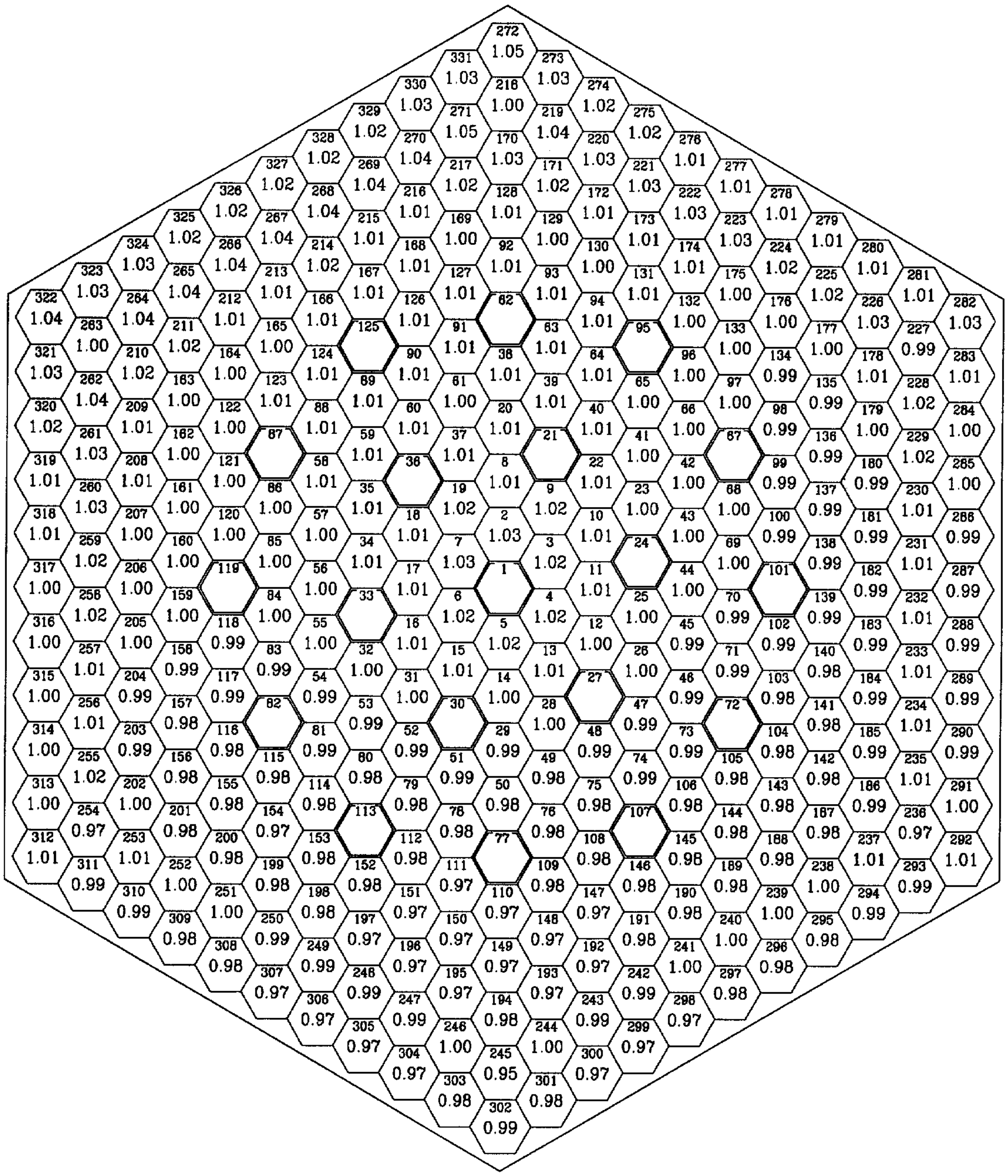


Fig.3.4. Pin-by-Pin Power Distribution in the Most Powered Assembly in BOC. Equilibrium Cycle for Uranium Reference Core with Boron BPRs



T	0.00	EFPD
W	3000.0	MW
$C_{H_3BO_3}$	5.66	g/kg
Ql_{max}	309.5	W/cm
Fuel assembly	19	
Level	4	
Fuel rod	271	
Kk_{max}	1.15	

Fig.3.5. Pin-by-Pin Power Distribution in the Most Powered Assembly in EOC. Equilibrium Cycle for Uranium Reference Core with Boron BPRs



T	286.88	EFPD
W	3000.0	MW
$C_{H_3BO_3}$	0.00	g/kg
Ql_{max}	253.5	W/cm
Fuel assembly	19	
Level	2	
Fuel rod	272	
Kk_{max}	1.05	

Figure 3.6. Control Rods Grouping and Positions of In-core Self-Powered Detectors

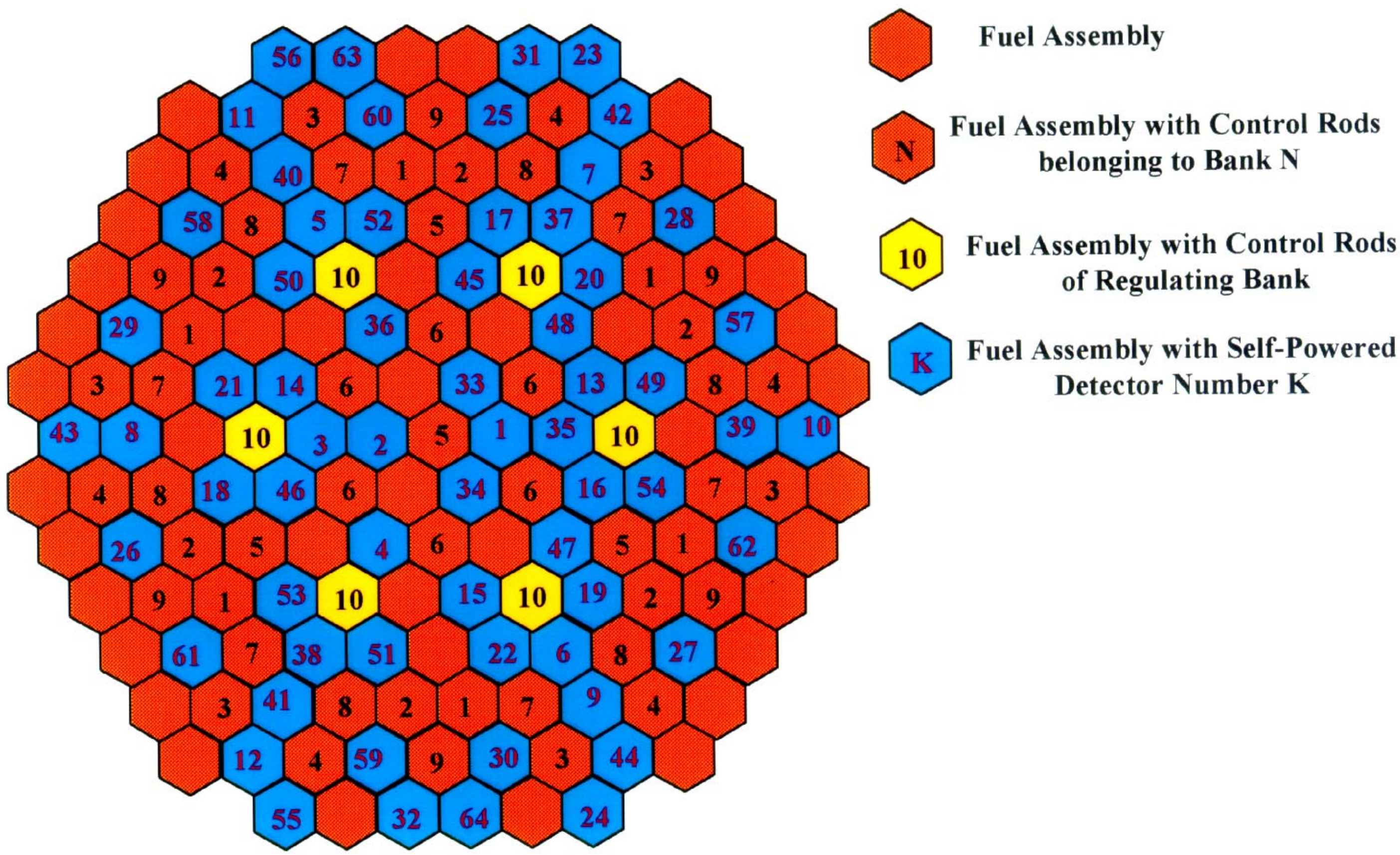


Fig.3.7. Reloading Scheme.
First Cycle with 3 MOX LTAs

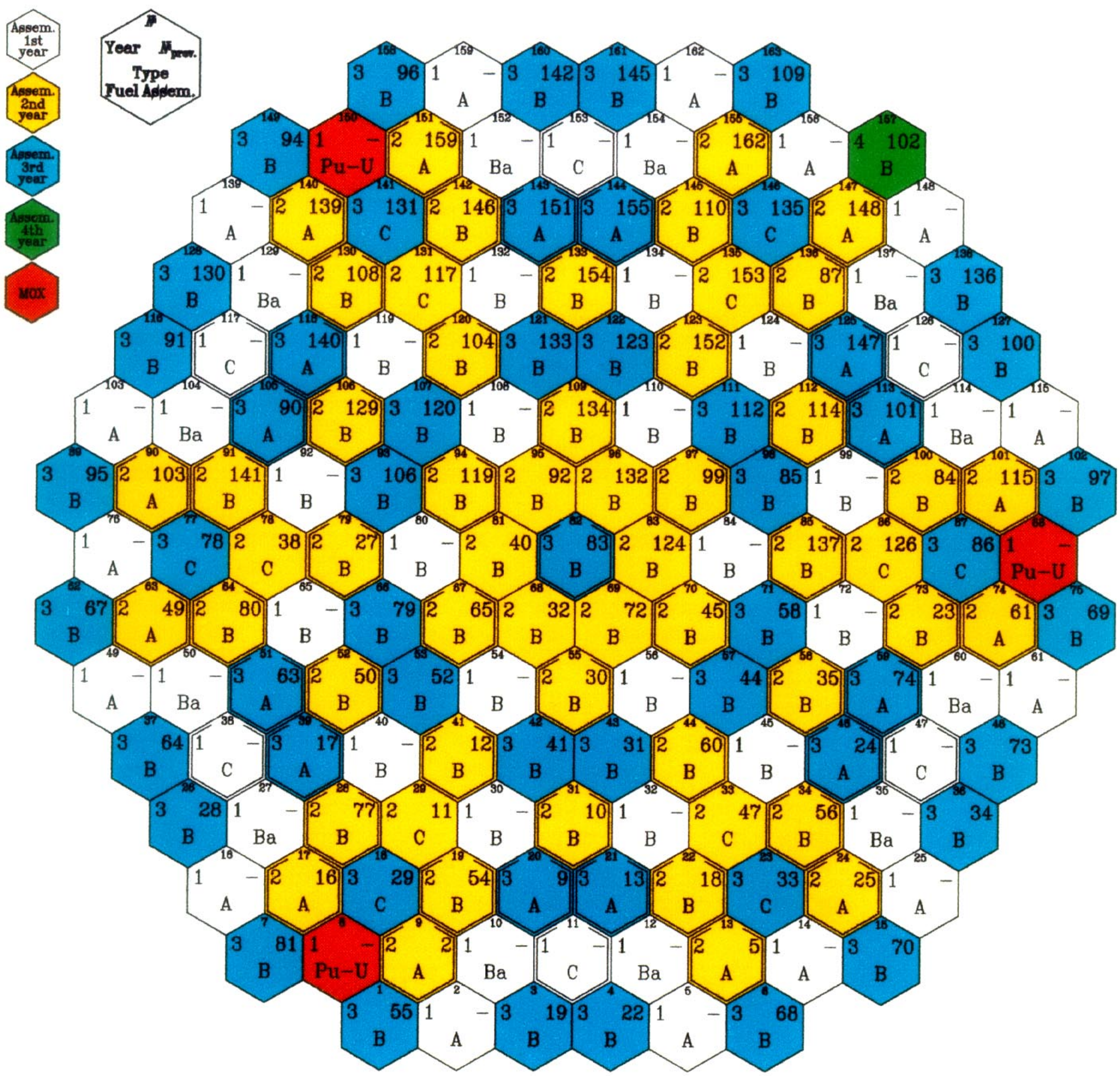


Fig.3.8. Assembly-by-Assembly Power Distribution.
First Cycle with 3 MOX LTAs of «Island-2» Type (Pu3.8-2.8, U-3.7)

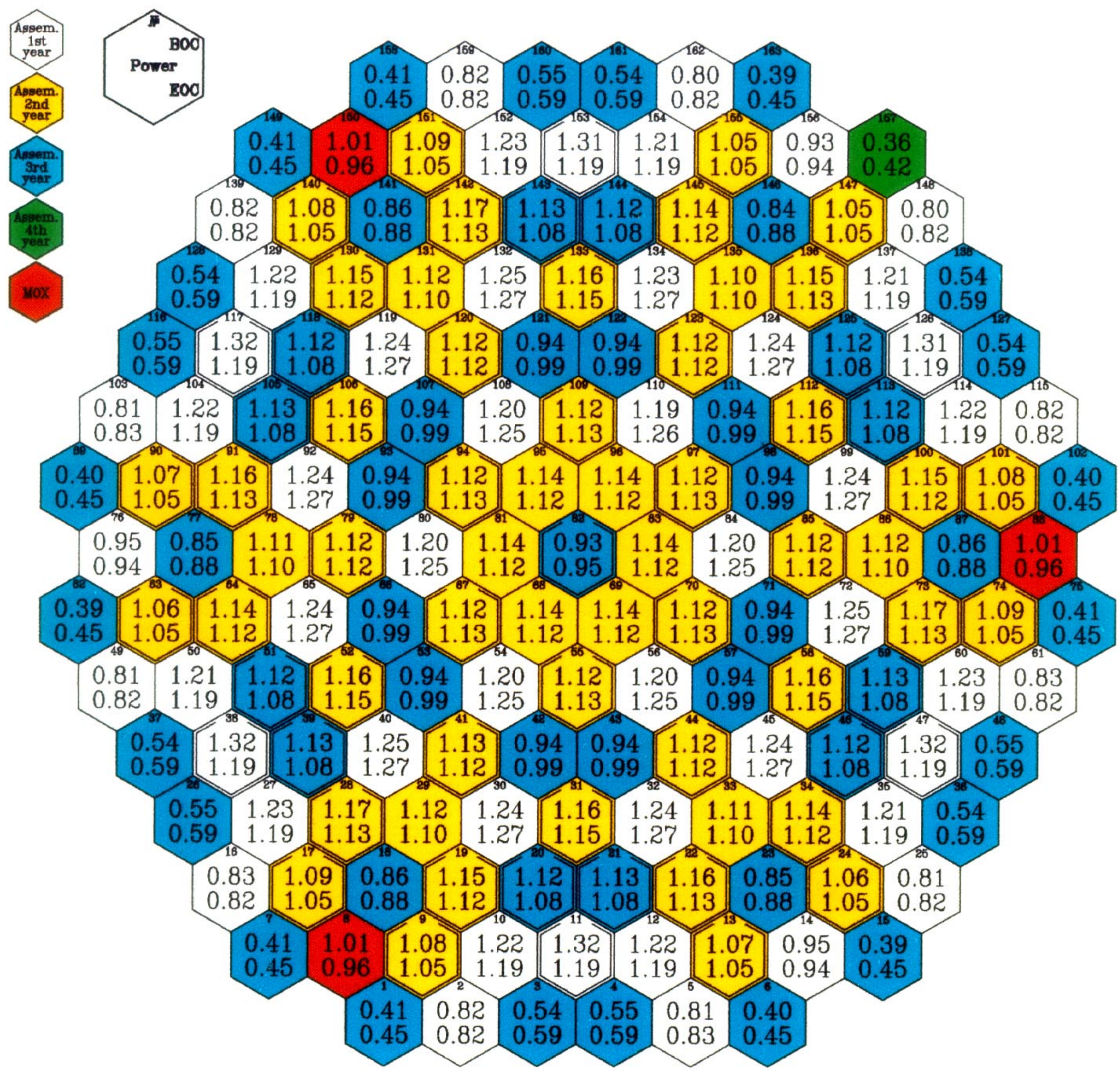


***Fig.3.9. Assembly-by-Assembly Burnup Distribution.
First Cycle with 3 MOX LTAs of «Island-2» Type (Pu3.8-2.8, U-3.7)***

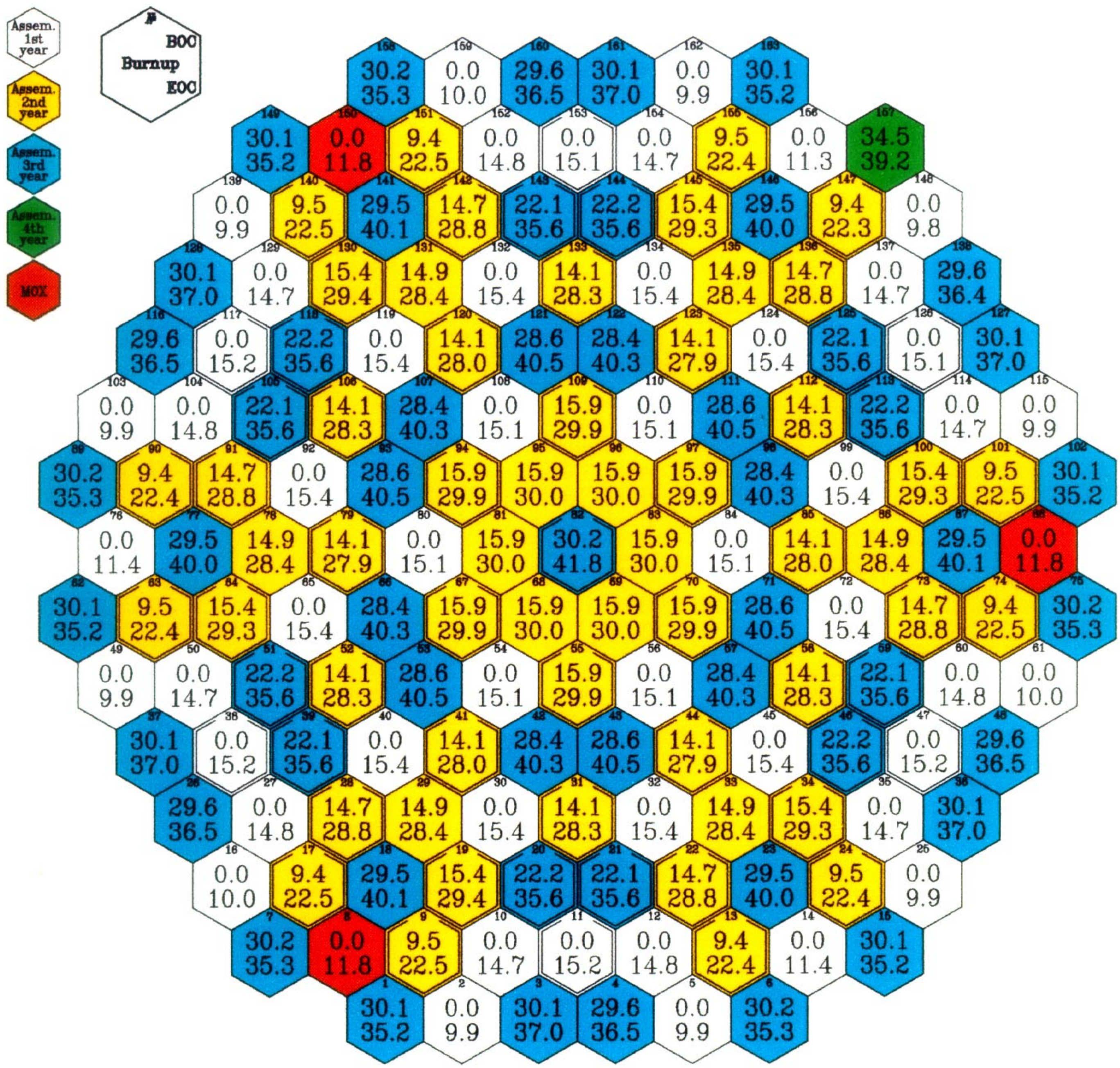


***Fig.3.10. Assembly-by-Assembly Temperature Drop Distribution.
First Cycle with 3 MOX LTAs of «Island-2» Type (Pu3.8-2.8, U-3.7)***

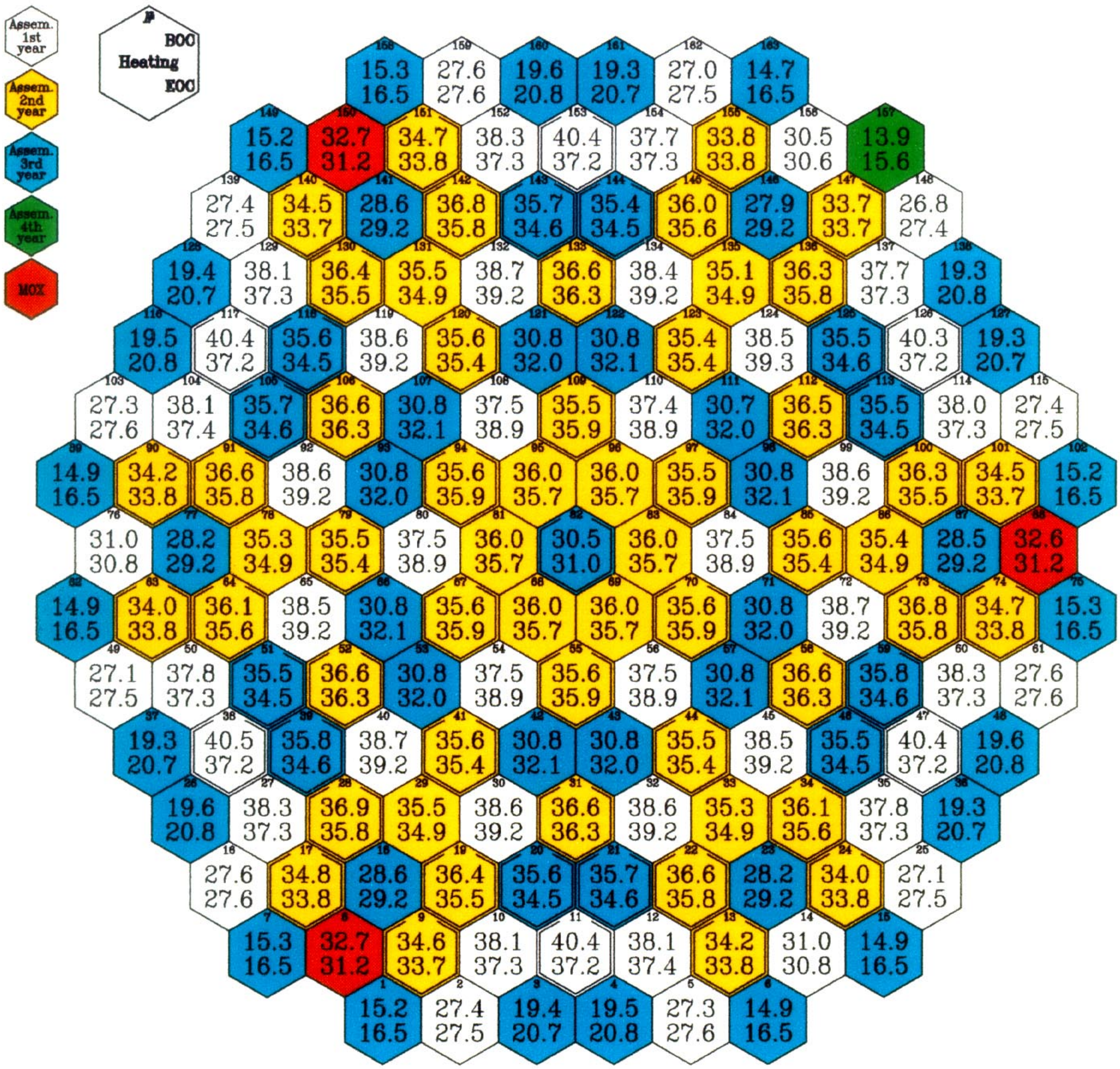
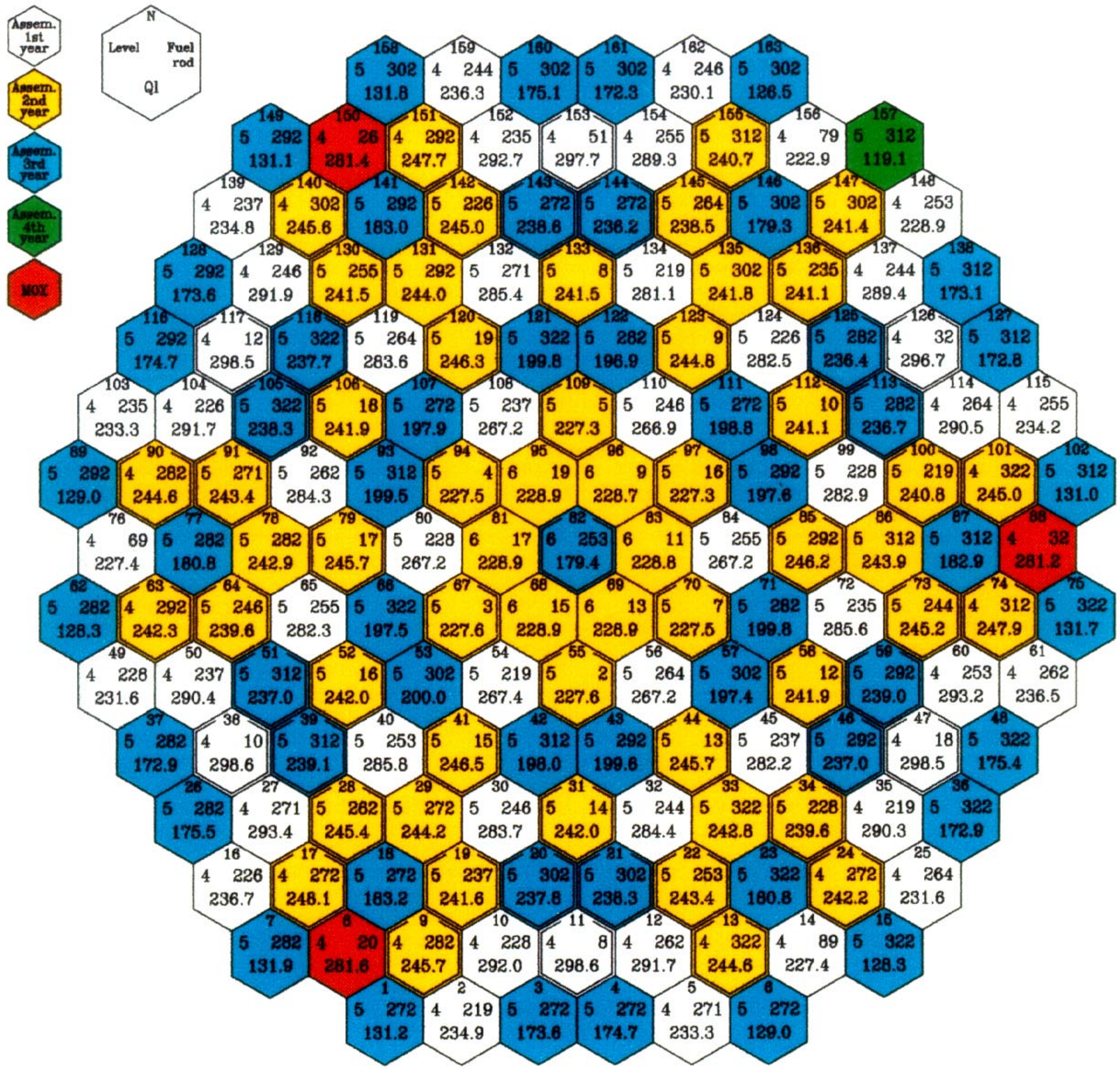
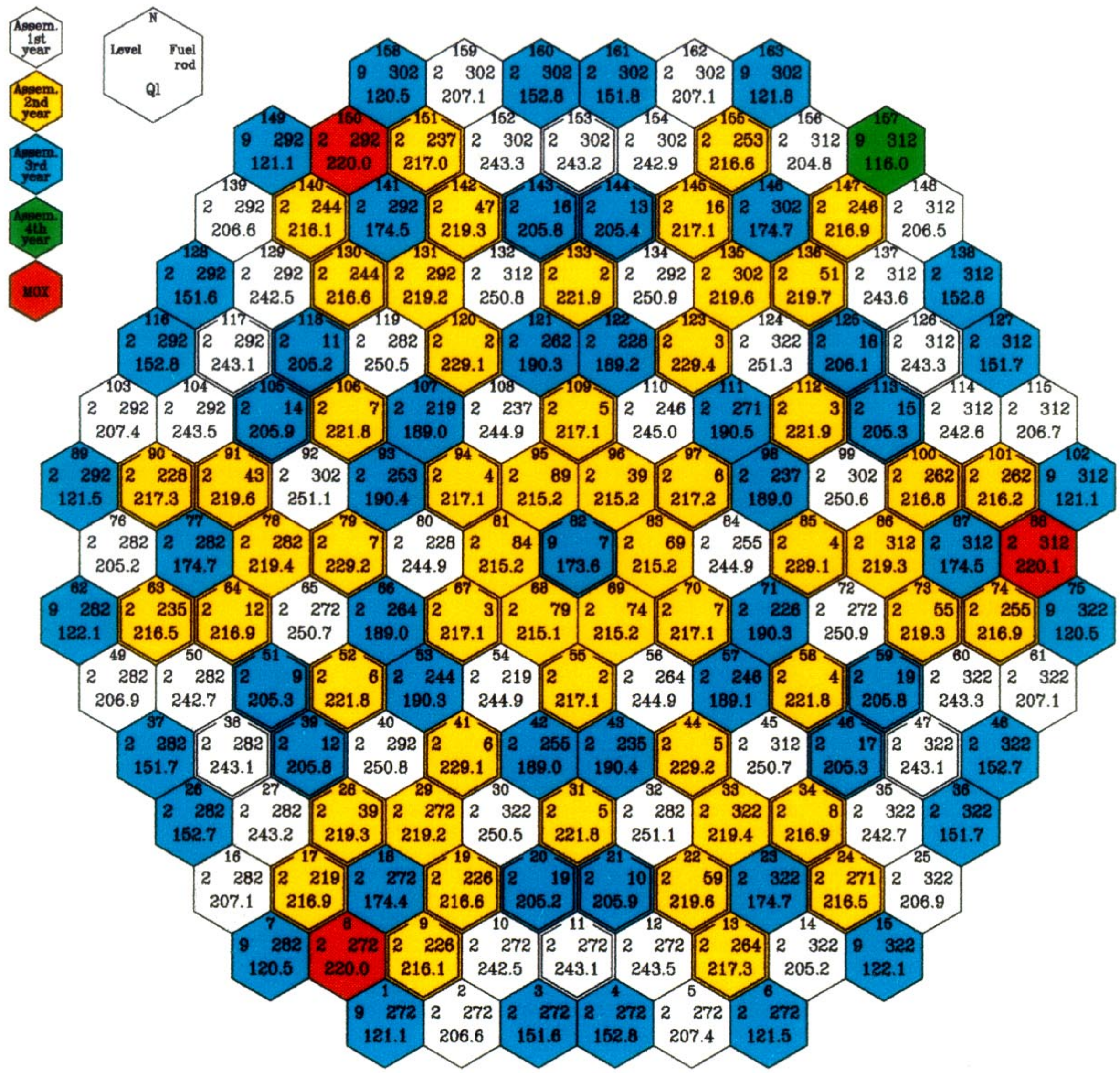


Fig.3.11. Assembly-by-Assembly Maximum Linear Power Distribution in BOC. First Cycle with 3 MOX LTAs of «Island-2» Type (Pu3.8-2.8-U3.7)



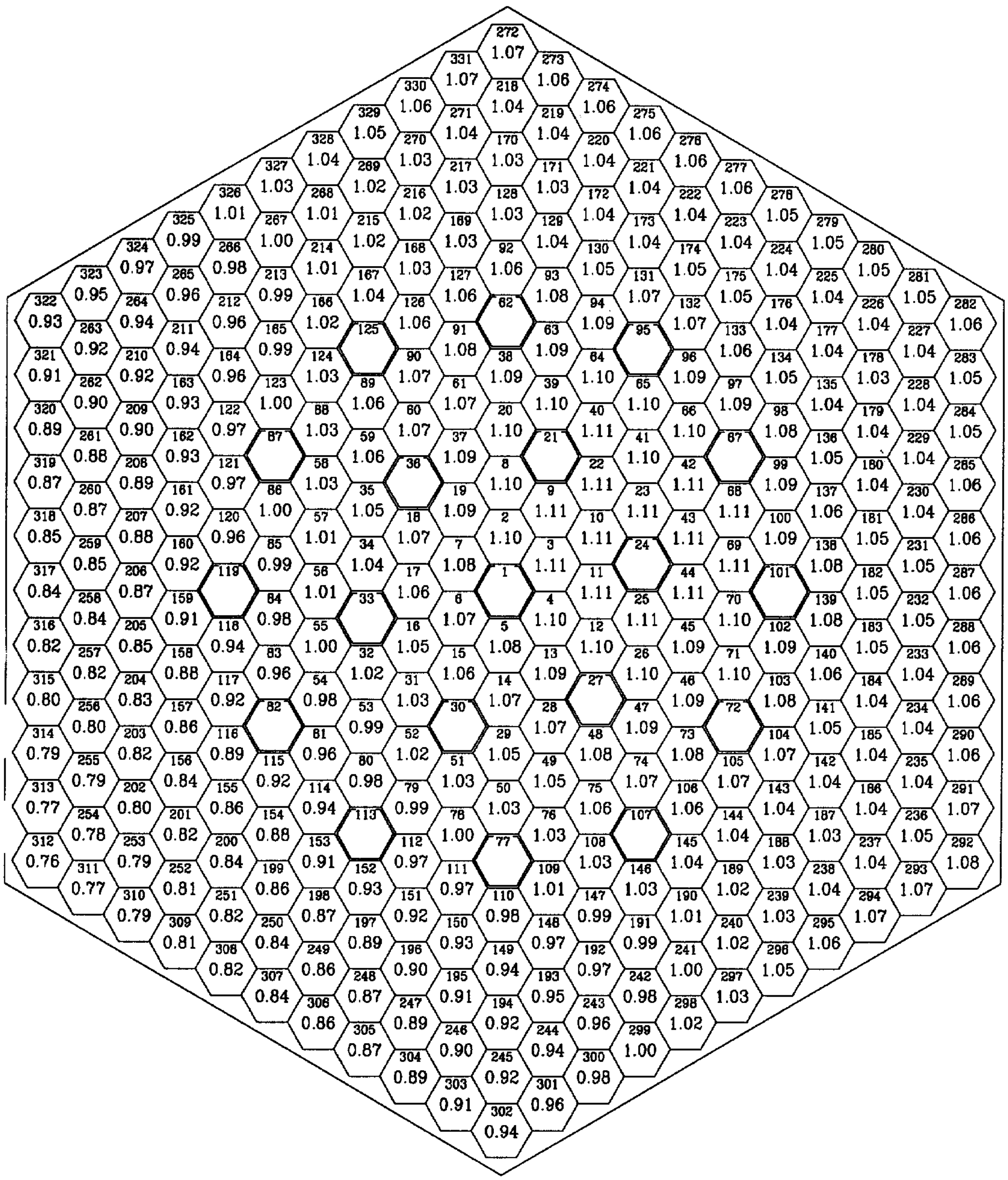
T	=	0.00 EFPD
W	=	3000.0 MW
$C_{H_3BO_3}$	=	5.77 g/kg
Ql_{max}	=	298.6 W/cm
Fuel ass.	=	38
Level	=	4
Fuel rod	=	10

Fig.3.12. Assembly-by-Assembly Maximum Linear Power Distribution in EOC. First Cycle with 3 MOX LTAs 100%Pu (Pu3.8-2.8-U3.7)



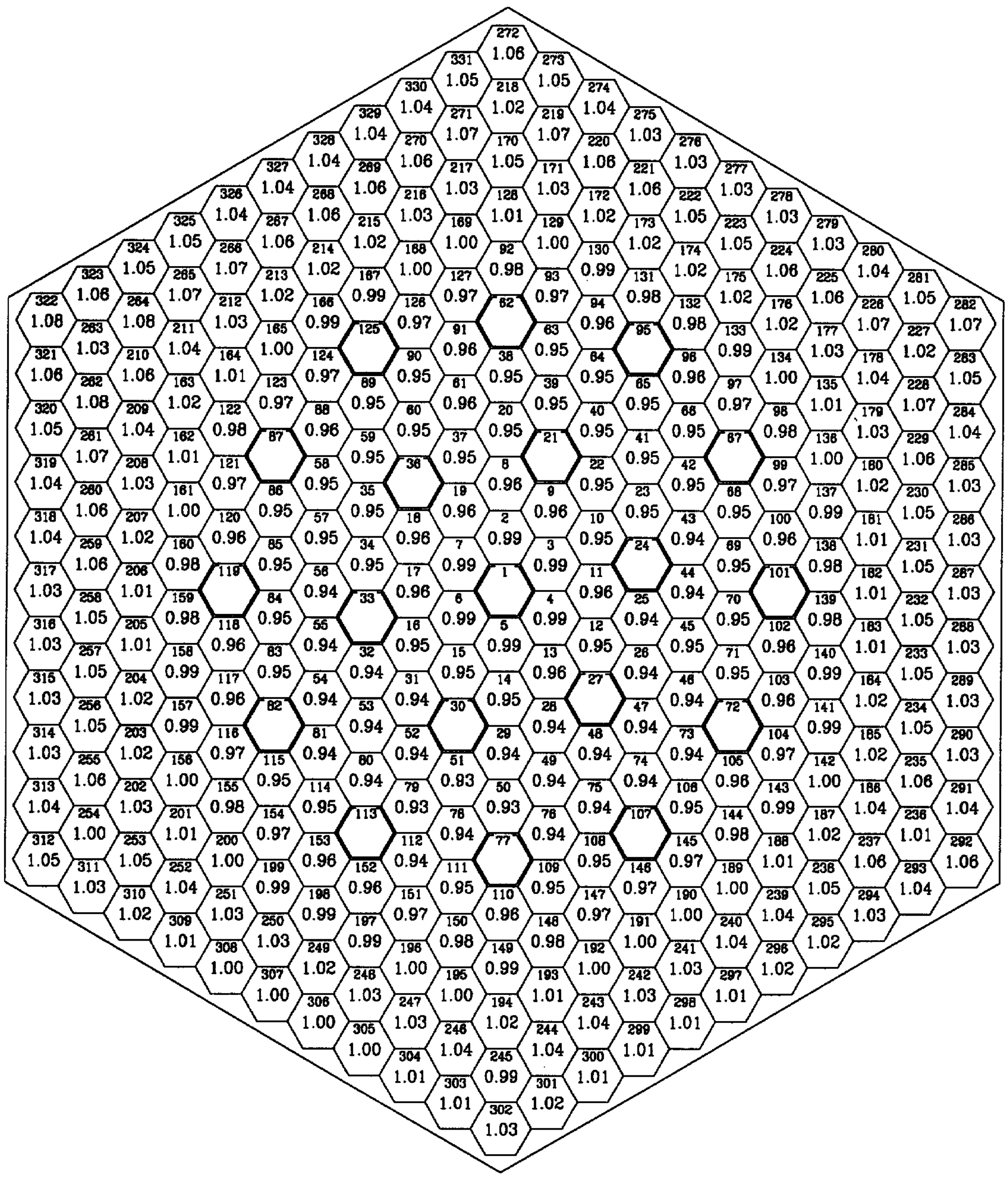
T	=	287.40 EFPD
W	=	3000.0 MW
$C_{H_3BO_3}$	=	0.00 g/kg
Ql_{max}	=	251.3 W/cm
Fuel ass.	=	124
Level	=	2
Fuel rod	=	322

Fig.3.13. Pin-by-Pin Power Distribution in the Most Powered Assembly in BOC. First Cycle with 3 MOX LTAs of «Island-2» Type (Pu3.8-2.8, U-3.7)



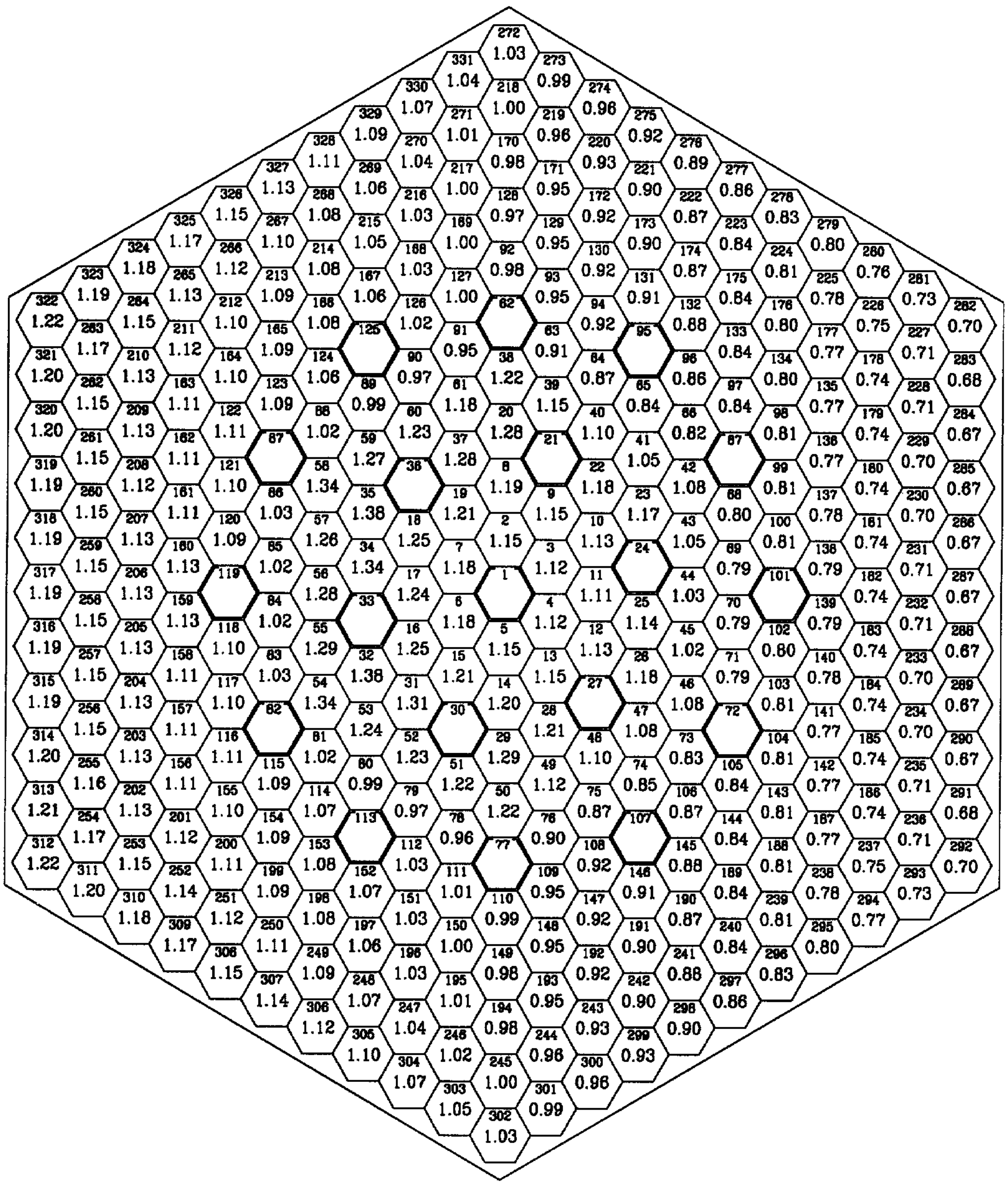
T	0.00	EFPD
W	3000.0	MW
$C_{H_3BO_3}$	5.77	g/kg
Ql	298.6	W/cm
Fuel assembly	38	
Level	4	
Fuel rod	10	
Kk_{max}	1.11	

Fig.3.14. Pin-by-Pin Power Distribution in the Most Powered Assembly in EOC. First Cycle with 3 MOX LTAs of «Island-2» Type (Pu3.8-2.8, U-3.7)



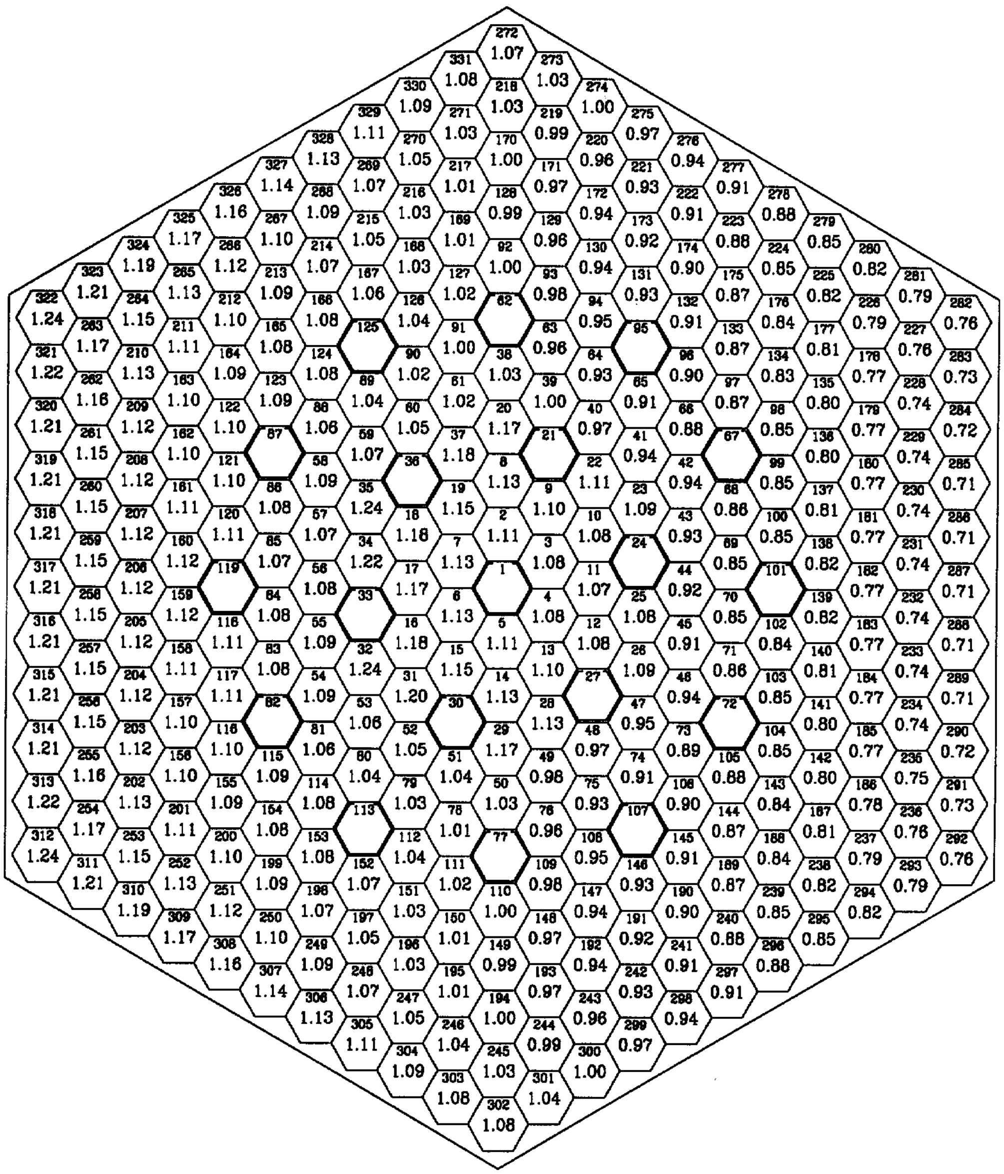
T	287.40	EFPD
W	3000.0	MW
$C_{H_3BO_3}$	0.00	g/kg
Burnup	18.8	
Fuel assembly	124	
Level	4	
Fuel rod	264	
Kb_{max}	1.08	

Fig.3.15. Pin-by-Pin Power Distribution in MOX LTA in BOC. First Cycle with 3 MOX LTAs of «Island-2» Type (Pu3.8-2.8, U-3.7)



T	0.00	EFPD
W	3000.0	MW
$C_{H_3BO_3}$	5.77	g/kg
Ql	281.2	W/cm
Fuel assembly	88	
Level	4	
Fuel rod	32	
Kk_{max}	1.38	

Fig.3.16. Pin-by-Pin Power Distribution in MOX LTA in EOC. First Cycle with 3 MOX LTAs of «Island-2» Type (Pu3.8-2.8, U-3.7)



T	287.40	EFPD
W	3000.0	MW
$C_{H_3BO_3}$	0.00	g/kg
Ql	208.7	W/cm
Fuel assembly	88	
Level	4	
Fuel rod	312	
Kk_{max}	1.24	

Fig.3.17. Reloading Scheme.
Second Cycle with 3 MOX LTAs

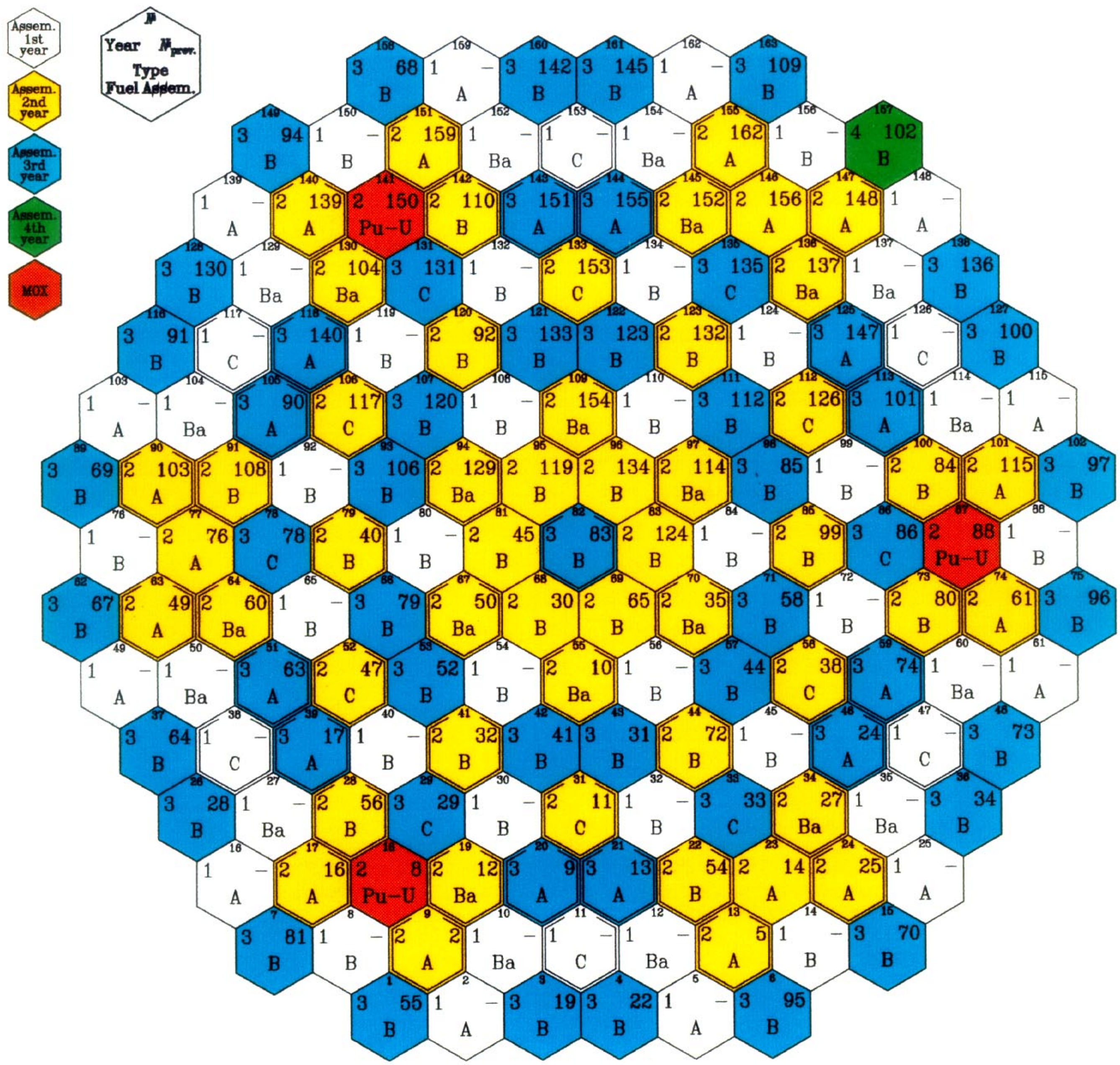


Fig.3.18. Assembly-by-Assembly Power Distribution.
Second Cycle with 3 MOX LTAs of «Island-2» Type (Pu3.8-2.8-U3.7)

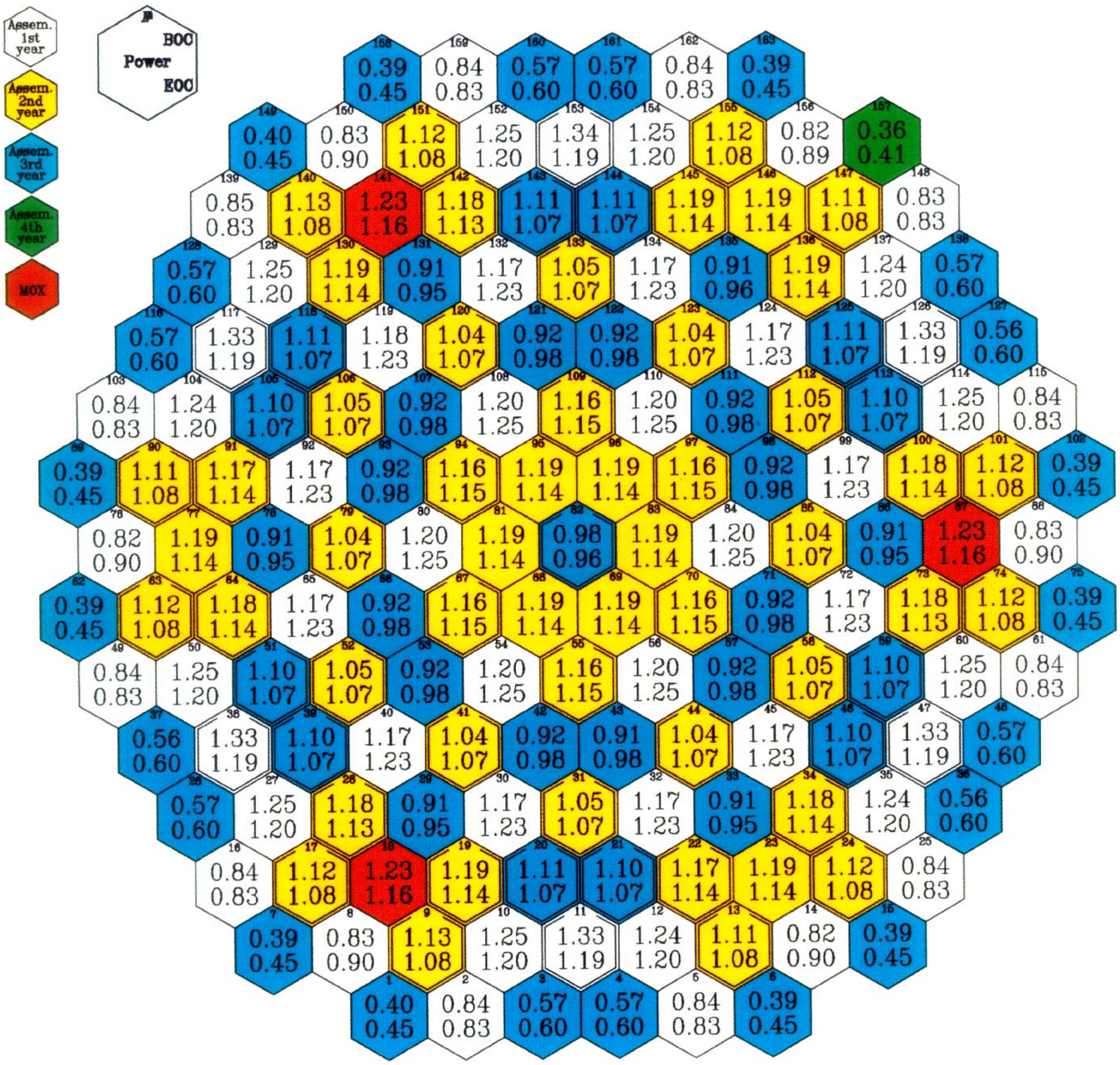


Fig.3.19. Assembly-by-Assembly Burnup Distribution.
Second Cycle with 3 MOX LTAs of «Island-2» Type (Pu3.8-2.8-U3.7)

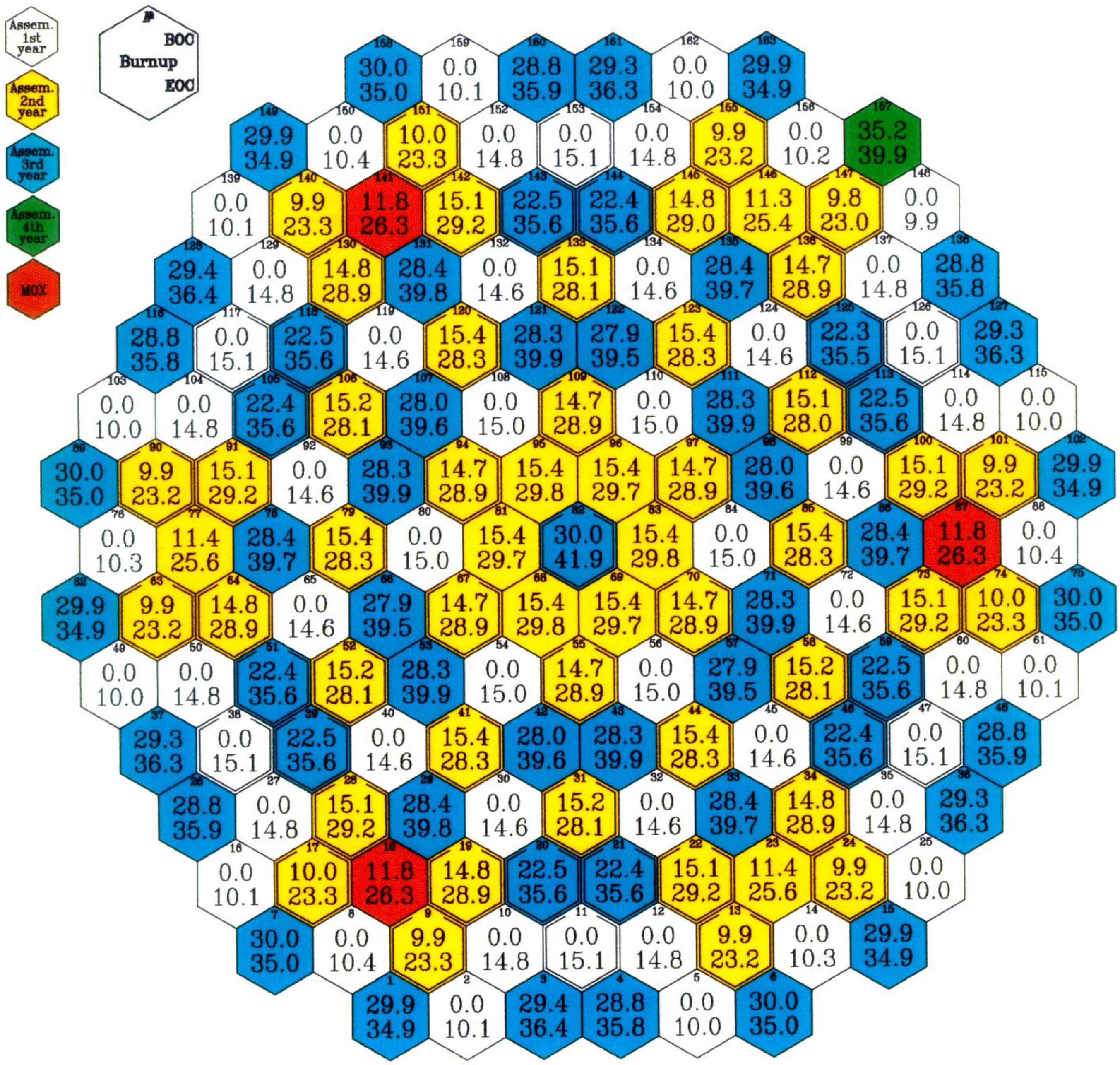


***Fig.3.20. Assembly-by-Assembly Temperature Drop Distribution.
Second Cycle with 3 MOX LTAs of «Island-2» Type (Pu3.8-2.8-U3.7)***

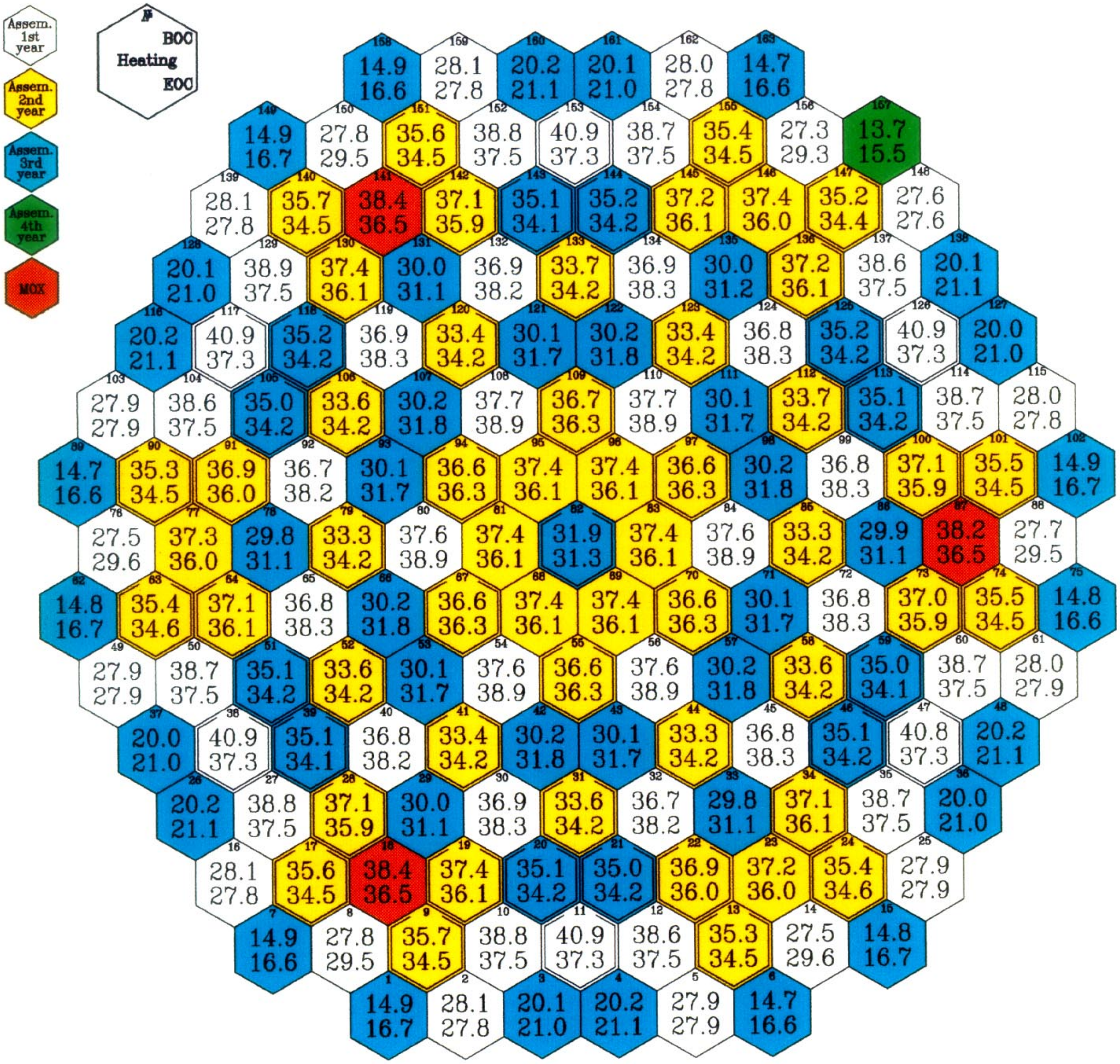
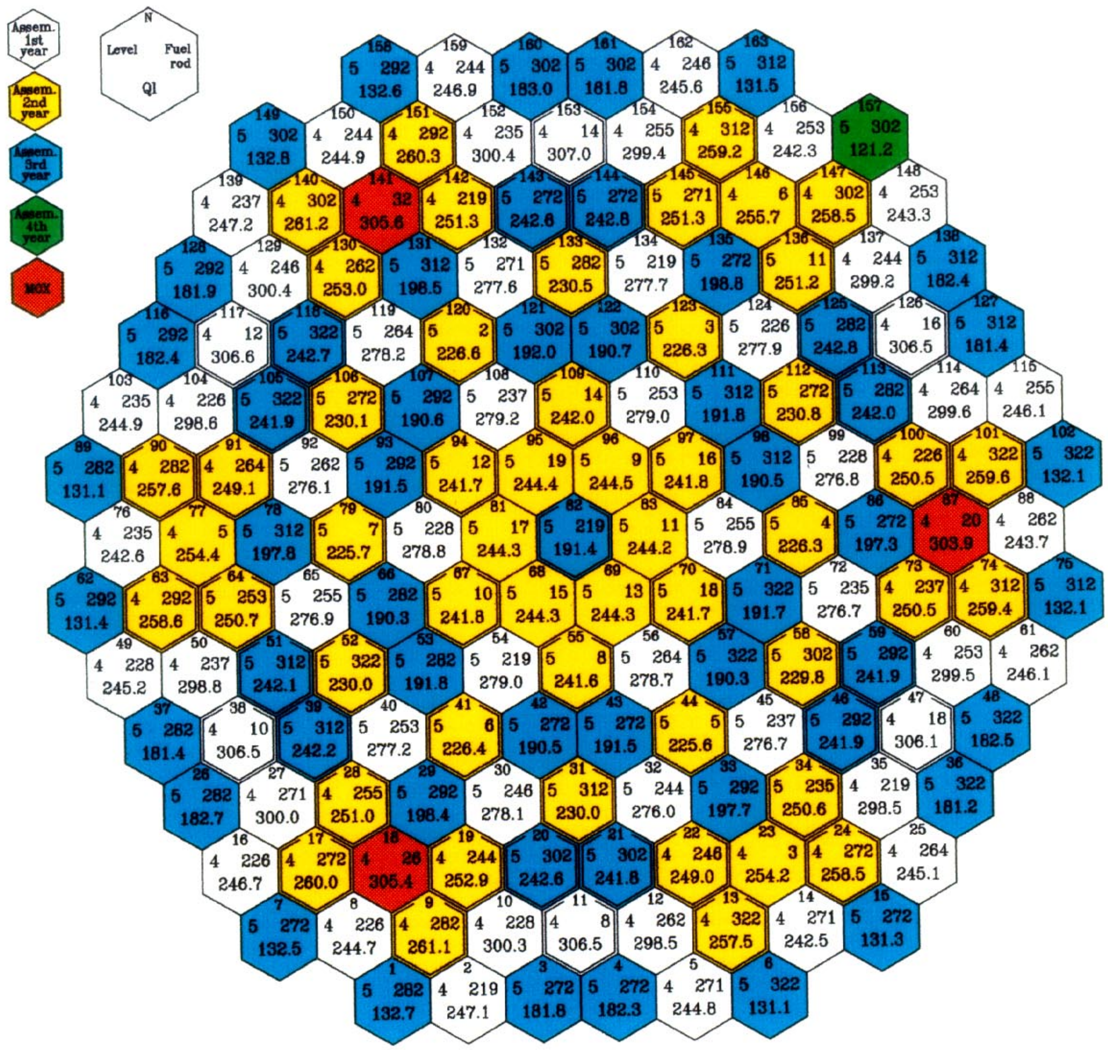
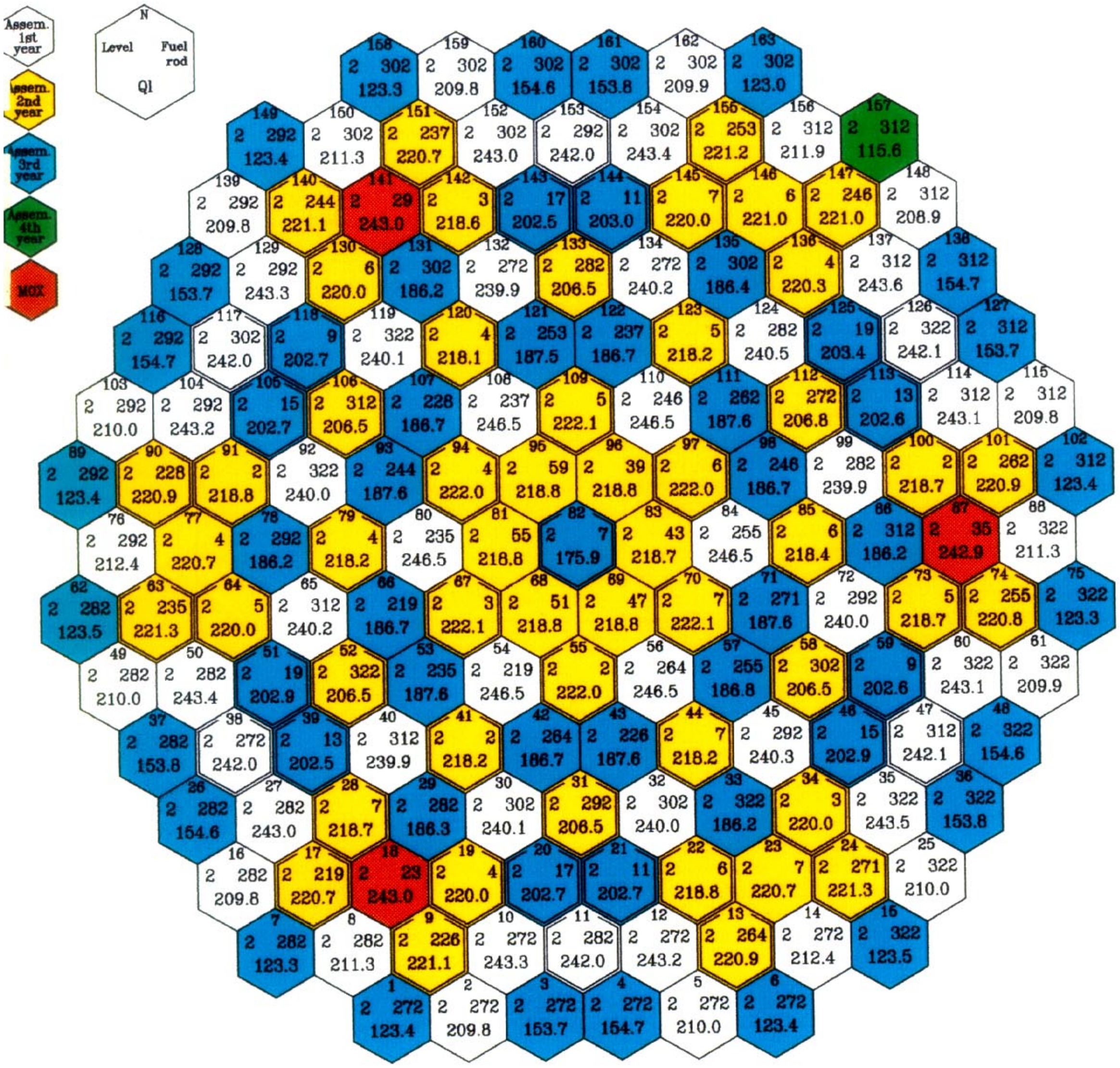


Fig.3.21. Assembly-by-Assembly Maximum Linear Pin Power Distribution in BOC. Second Cycle with 3 MOX LTAs of «Island-2» Type (Pu3.8-2.8-U3.7)



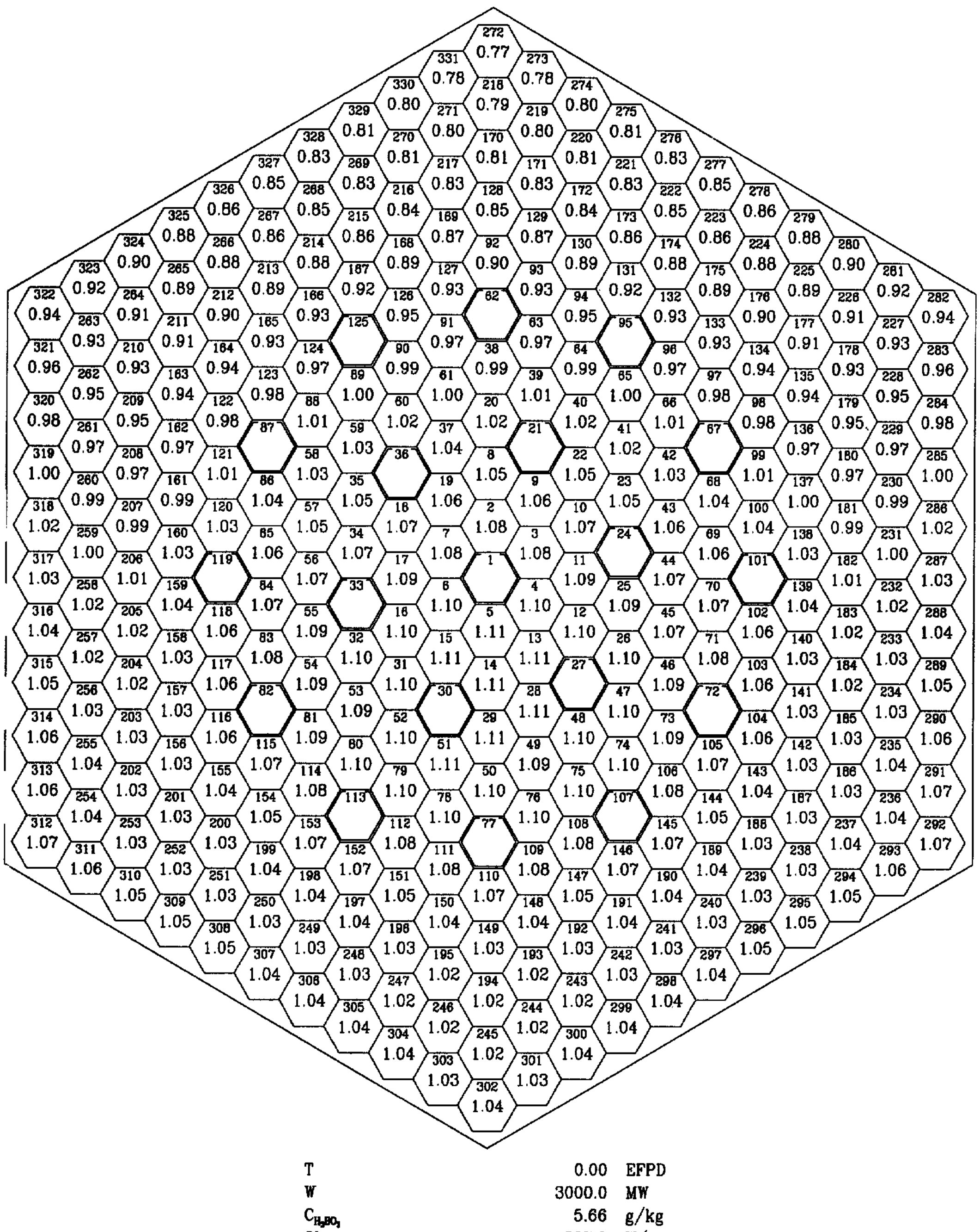
T	=	0.00 EFPD
W	=	3000.0 MW
$C_{H_3BO_3}$	=	5.66 g/kg
Ql_{max}	=	307.0 W/cm
Fuel ass.	=	153
Level	=	4
Fuel rod	=	14

Fig.3.22. Assembly-by-Assembly Maximum Linear Pin Power Distribution in EOC. Second Cycle with 3 MOX LTAs of «Island-2» Type (Pu3.8-2.8-U3.7)



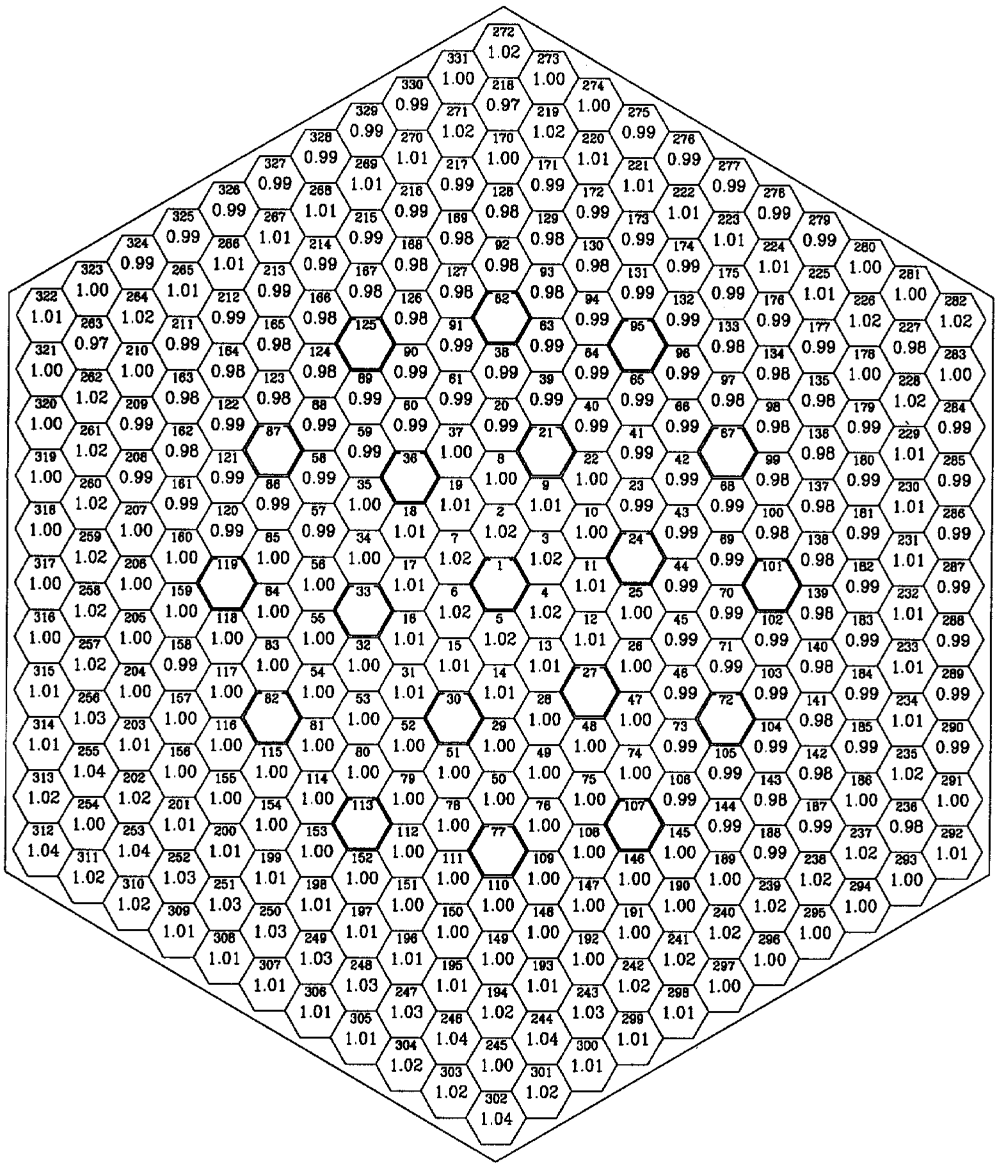
T	=	284.85 EFPD
W	=	3000.0 MW
$C_{H_3BO_3}$	=	0.00 g/kg
Ql_{max}	=	246.5 W/cm
Fuel ass.	=	56
Level	=	2
Fuel rod	=	264

Fig.3.23. Pin-by-Pin Power Distribution in the Most Powered Assembly in BOC. Second Cycle with 3 MOX LTAs of «Island-2» Type (Pu3.8-2.8-U3.7)



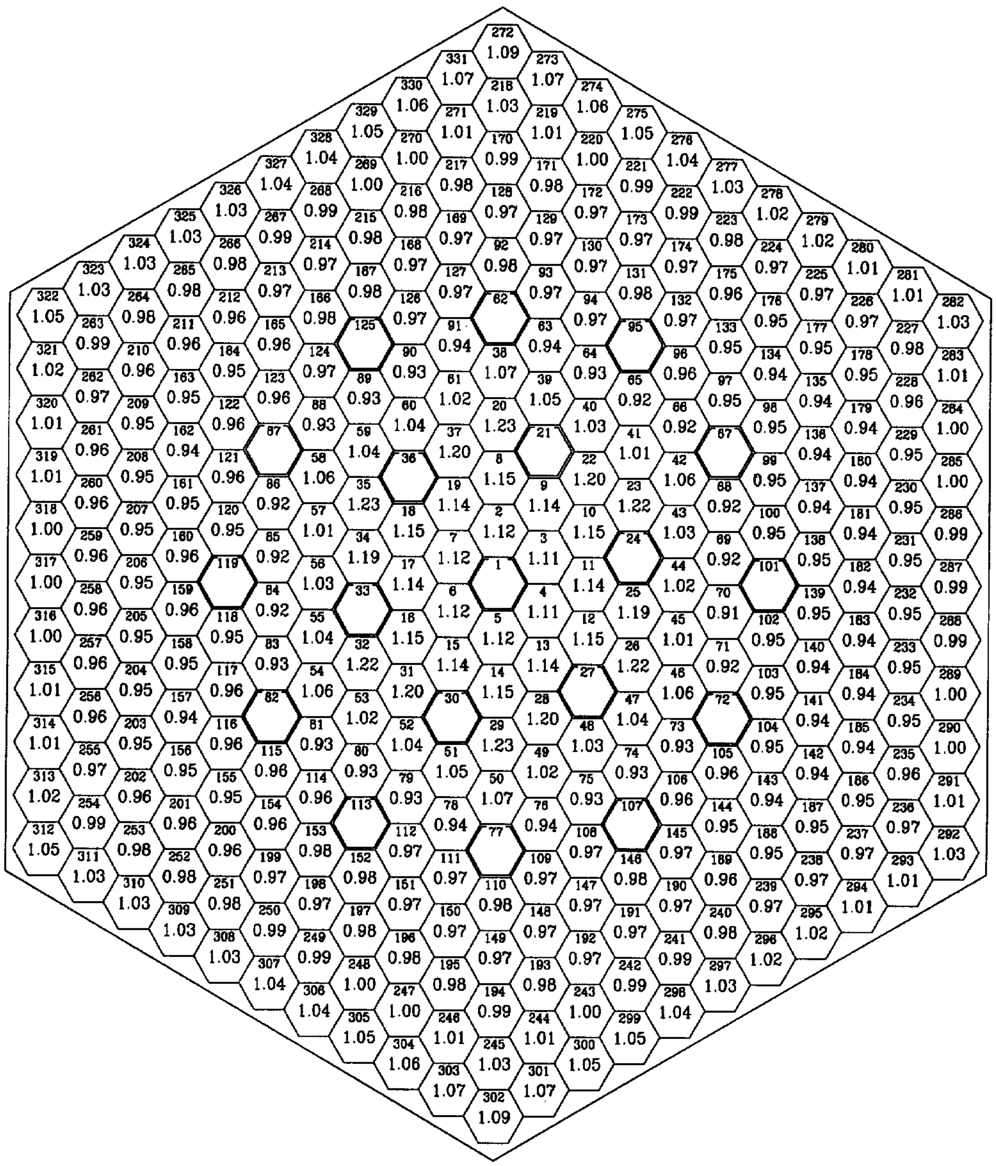
T	0.00	EFPD
W	3000.0	MW
$C_{H_3BO_3}$	5.66	g/kg
Ql	307.0	W/cm
Fuel assembly	153	
Level	4	
Fuel rod	14	
Kk_{max}	1.11	

Fig.3.24. Pin-by-Pin Power Distribution in the Most Powered Assembly in EOC. Second Cycle with 3 MOX LTAs of «Island-2» Type (Pu3.8-2.8-U3.7)



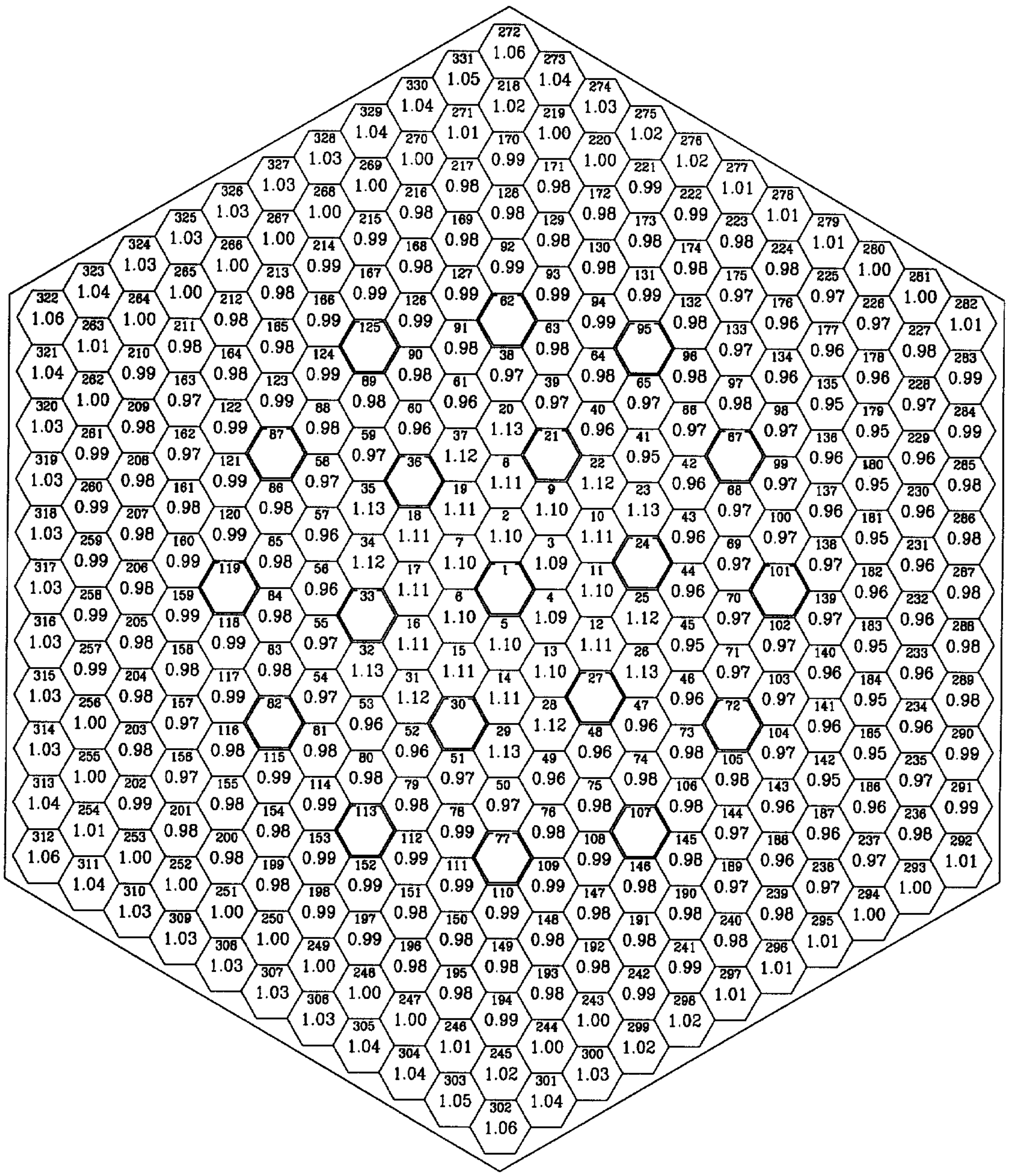
T	284.85	EFPD
W	3000.0	MW
$C_{H_3BO_3}$	0.00	g/kg
Ql	233.2	W/cm
Fuel assembly	110	
Level	4	
Fuel rod	246	
Kk_{max}	1.04	

Fig.3.25. Pin-by-Pin Power Distribution in MOX LTA in BOC. Second Cycle with 3 MOX LTAs of «Island-2» Type (Pu3.8-2.8-U3.7)



T	0.00	EFPD
W	3000.0	MW
$C_{H_3BO_3}$	5.66	g/kg
Ql	303.9	W/cm
Fuel assembly	87	
Level	4	
Fuel rod	20	
Kk_{max}	1.23	

Fig.3.26. Pin-by-Pin Power Distribution in MOX LTA in EOC. Second Cycle with 3 MOX LTAs of «Island-2» Type (Pu3.8-2.8-U3.7)



T	284.85	EFPD
W	3000.0	MW
$C_{H_3BO_3}$	0.00	g/kg
Ql	228.3	W/cm
Fuel assembly	87	
Level	4	
Fuel rod	35	
Kk_{max}	1.13	

Fig.3.27. Reloading scheme.
Third Cycle with 3 MOX LTAs

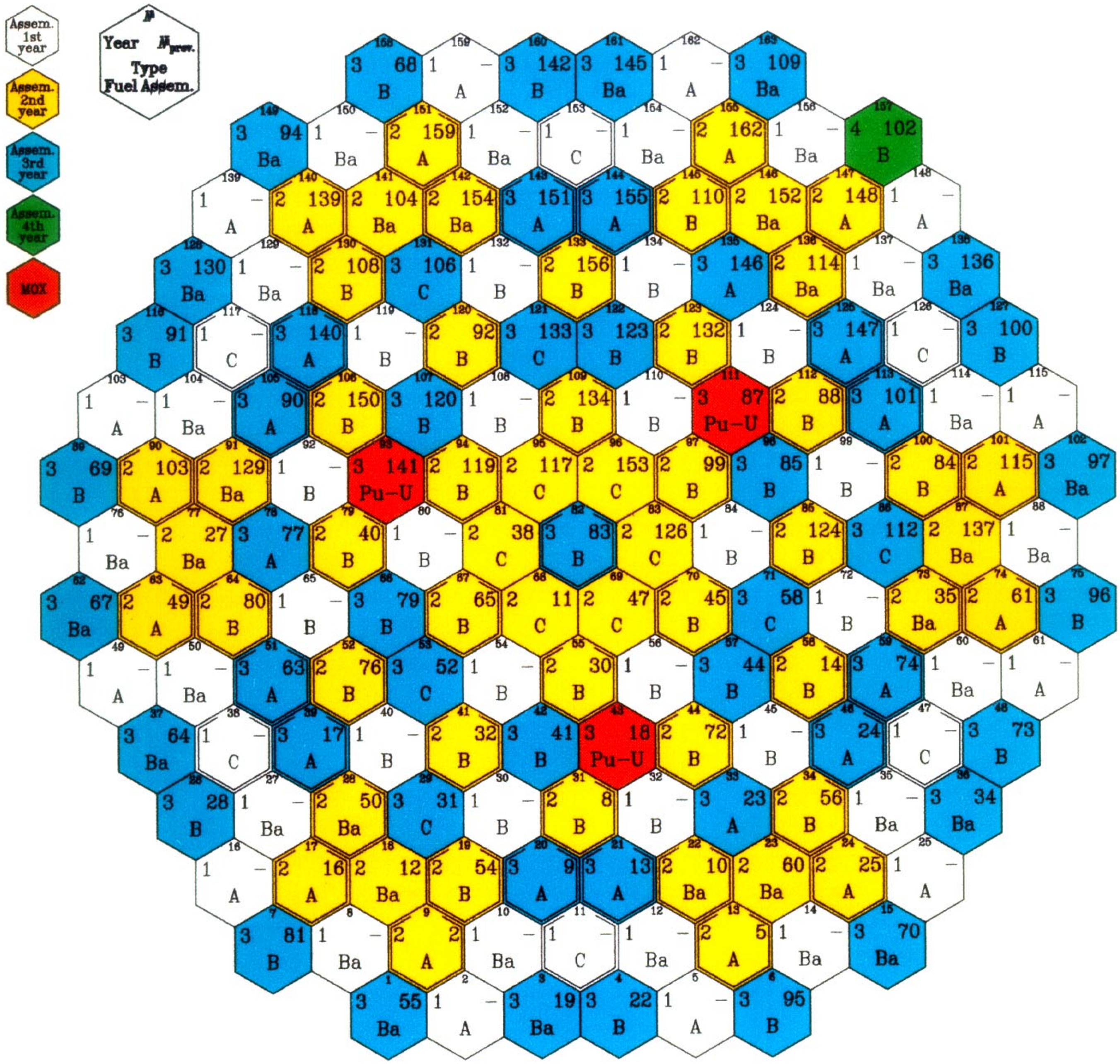


Fig.3.28. Assembly-by-Assembly Power Distribution.
Third Cycle with 3 MOX LTAs of "Island-2" Type (Pu3.8-2.8-U3.7)

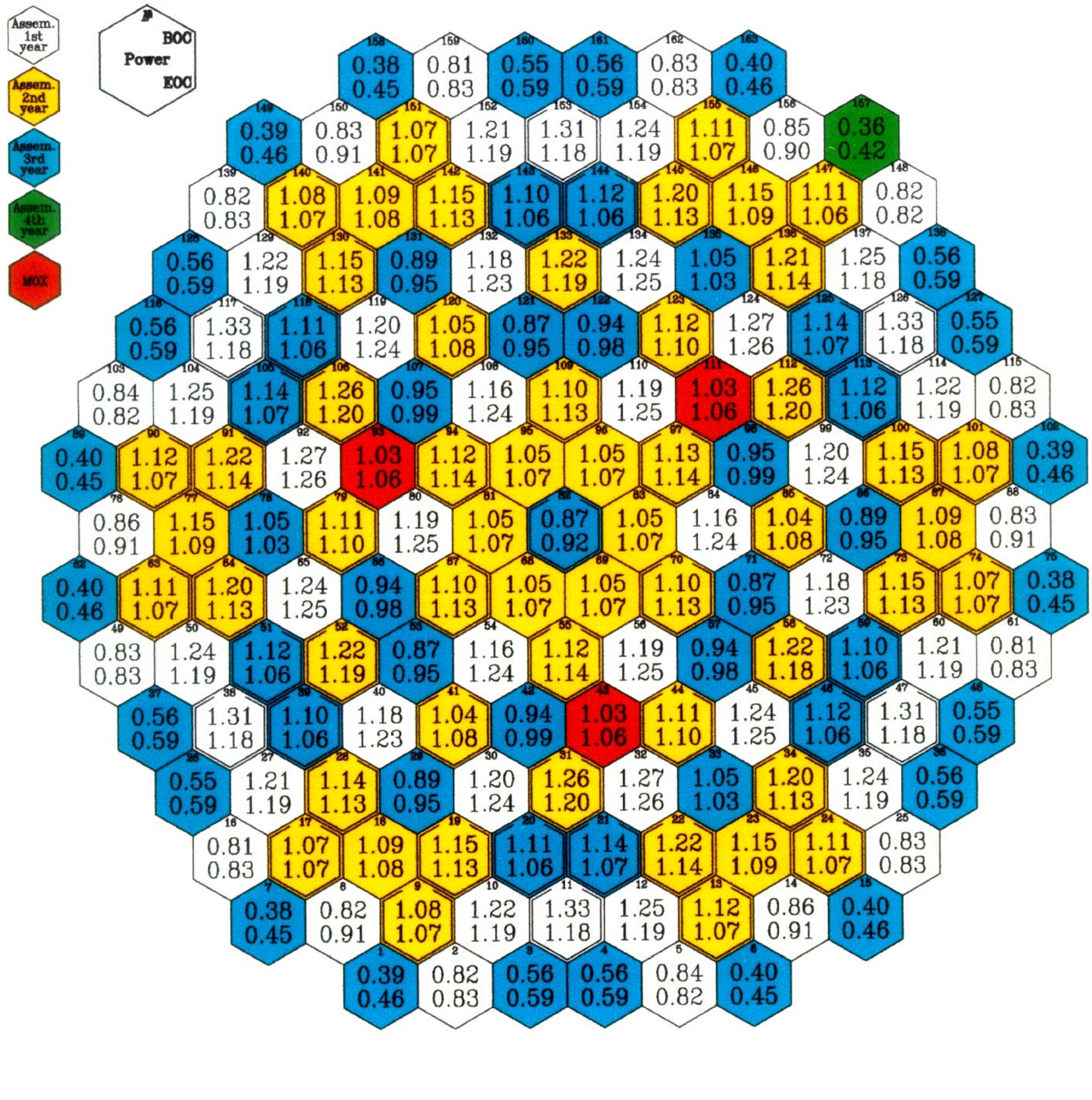


Fig.3.29. Assembly-by-Assembly Burnup Distribution.
Third Cycle with 3 MOX LTAs of "Island-2" Type (Pu3.8-2.8-U3.7)

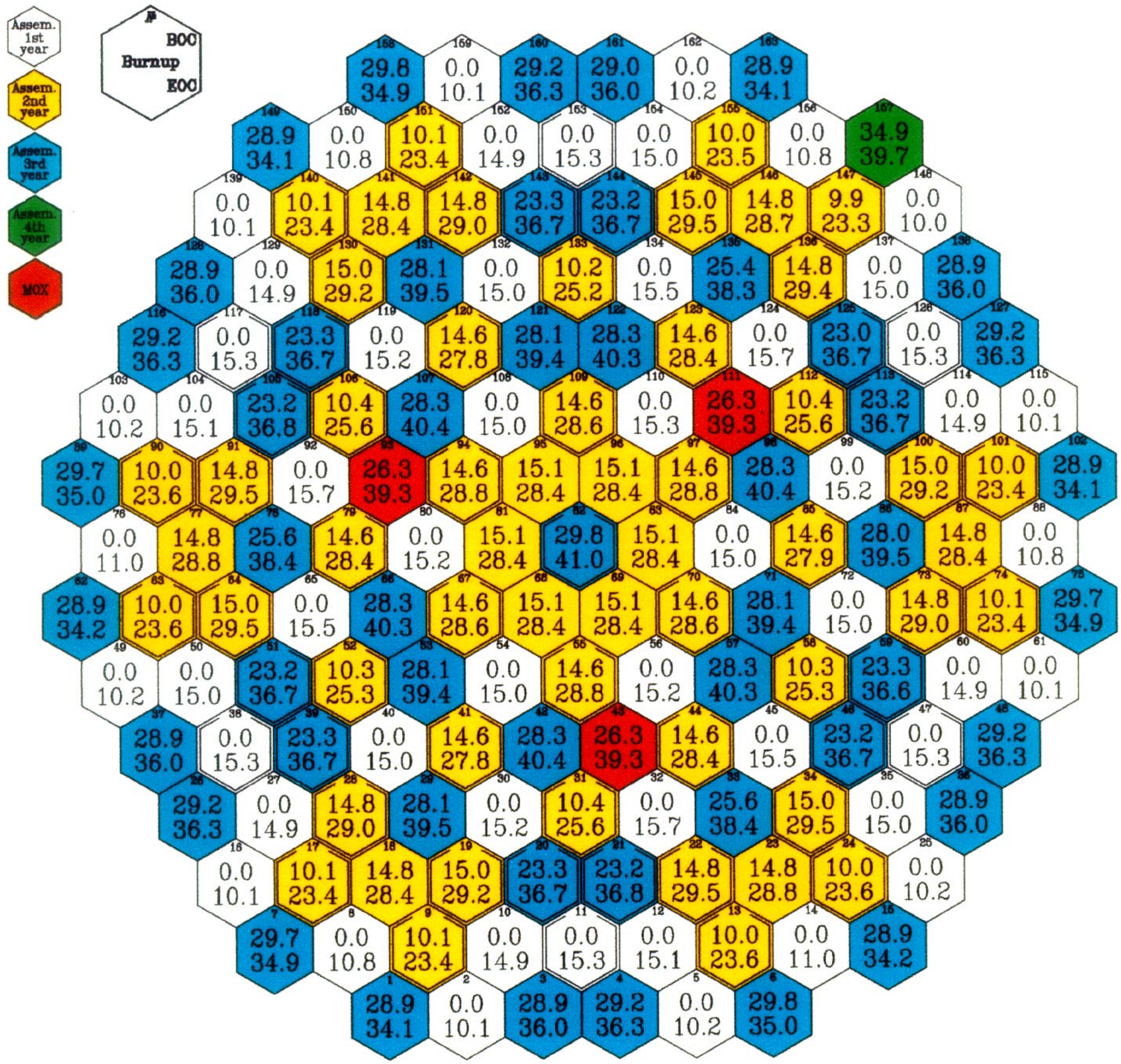


Fig.3.30. Assembly-by-Assembly Temperature Drop Distribution. Third Cycle with 3 MOX LTAs of «Island-2» Type (Pu3.8-2.8-U3.7)

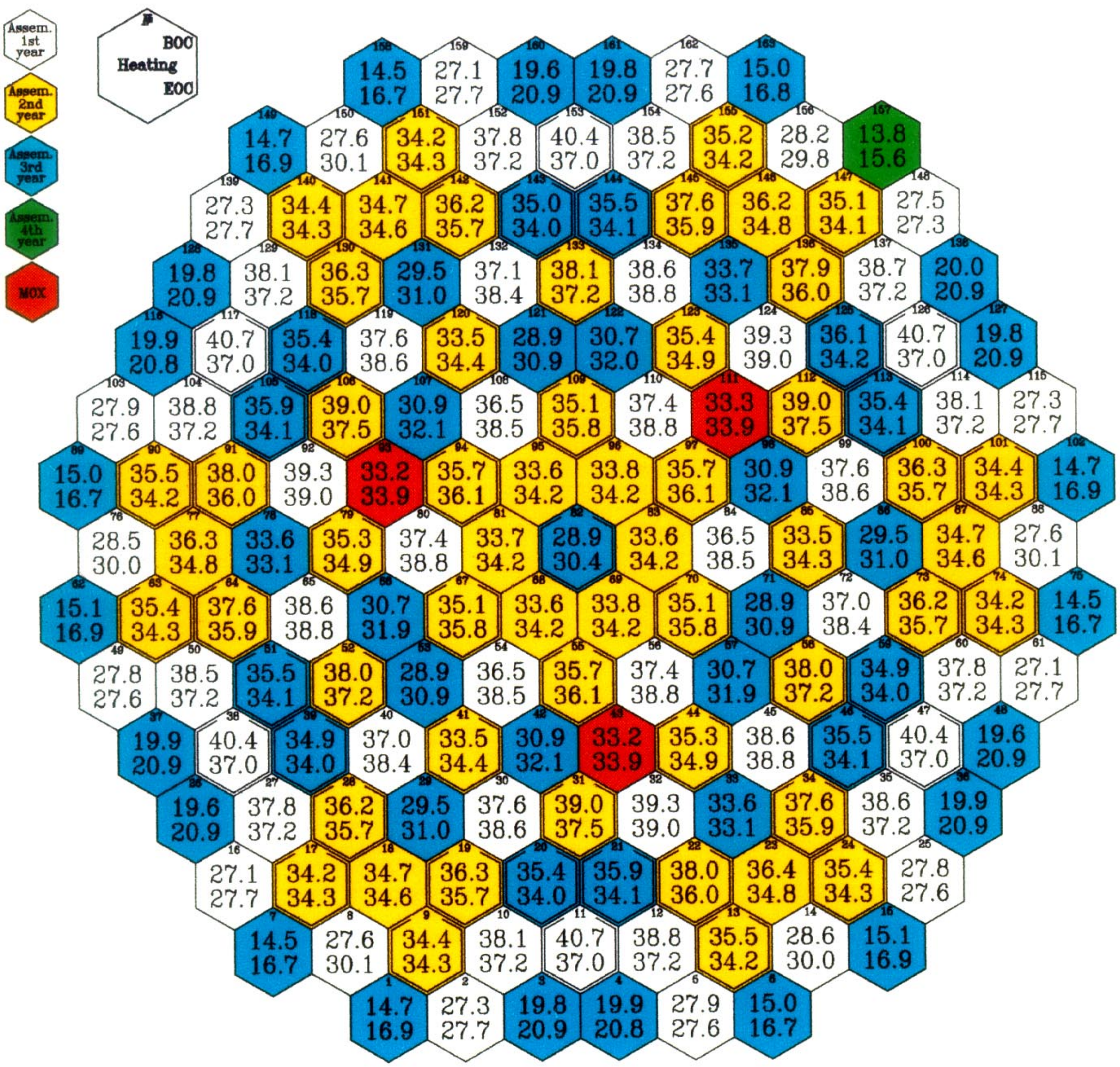
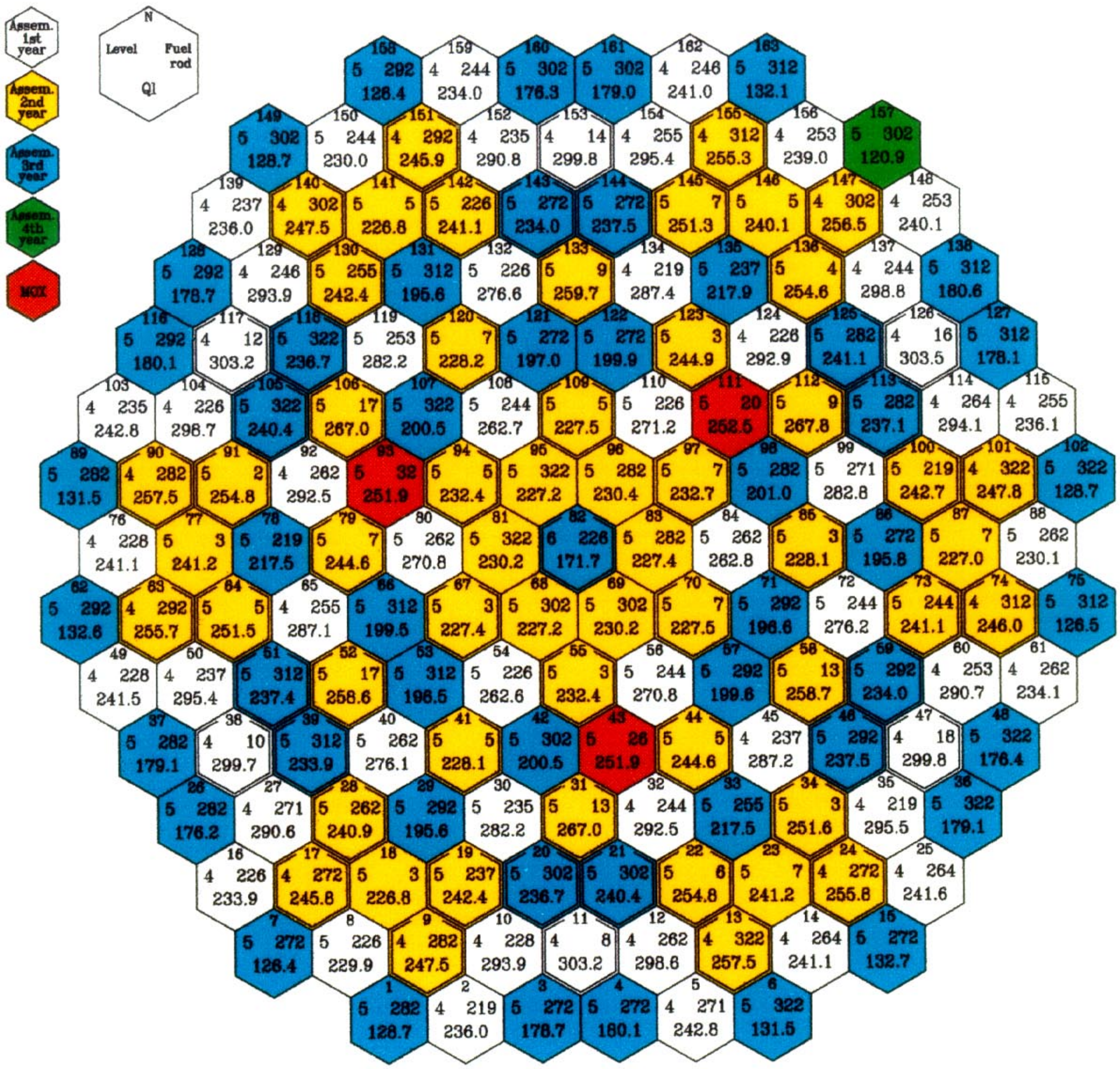
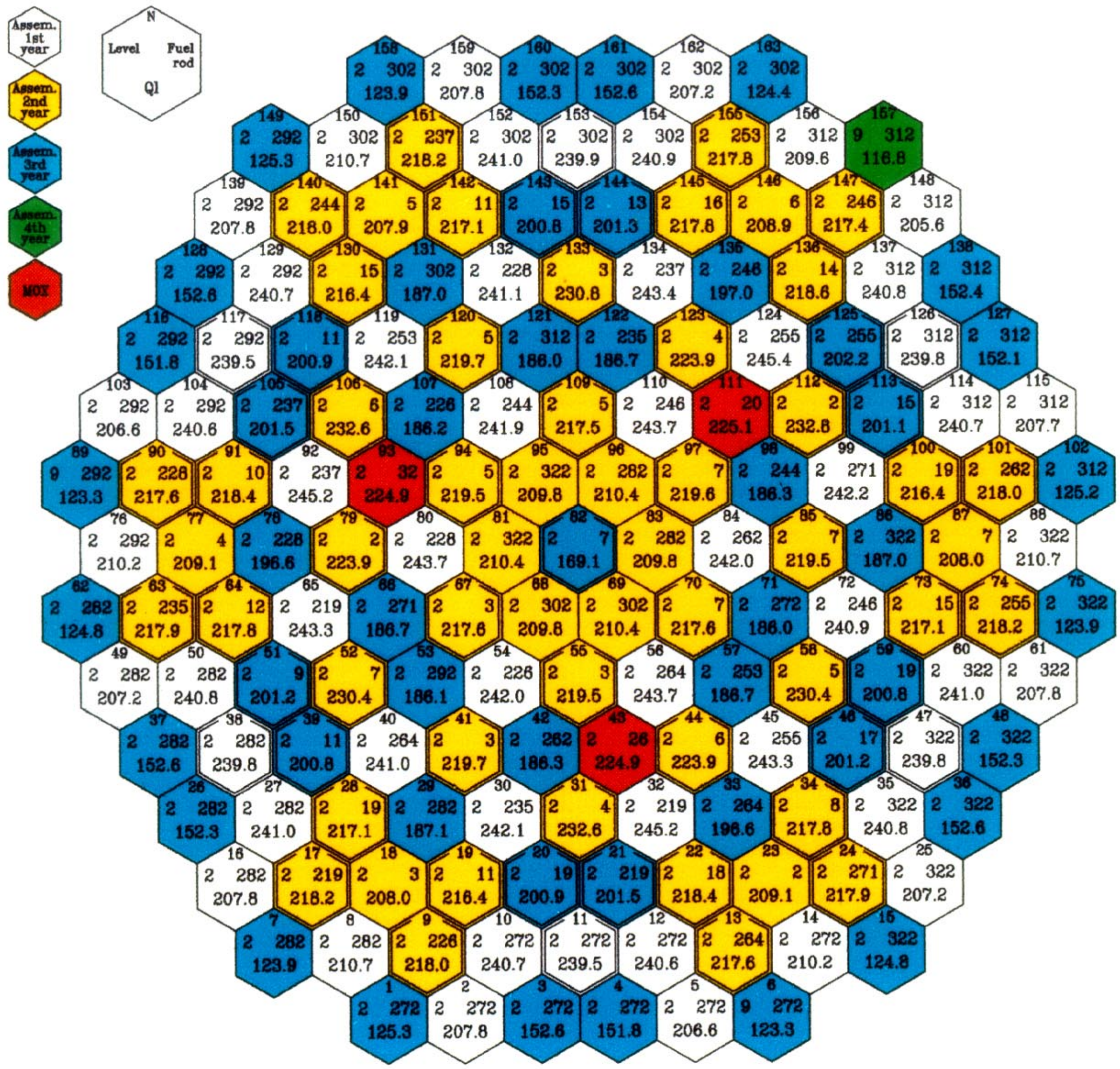


Fig.3.31. Assembly-by-Assembly Maximum Linear Power Distribution in BOC. Third Cycle with 3 MOX LTAs of «Island-2» Type (Pu3.8-2.8-U3.7)



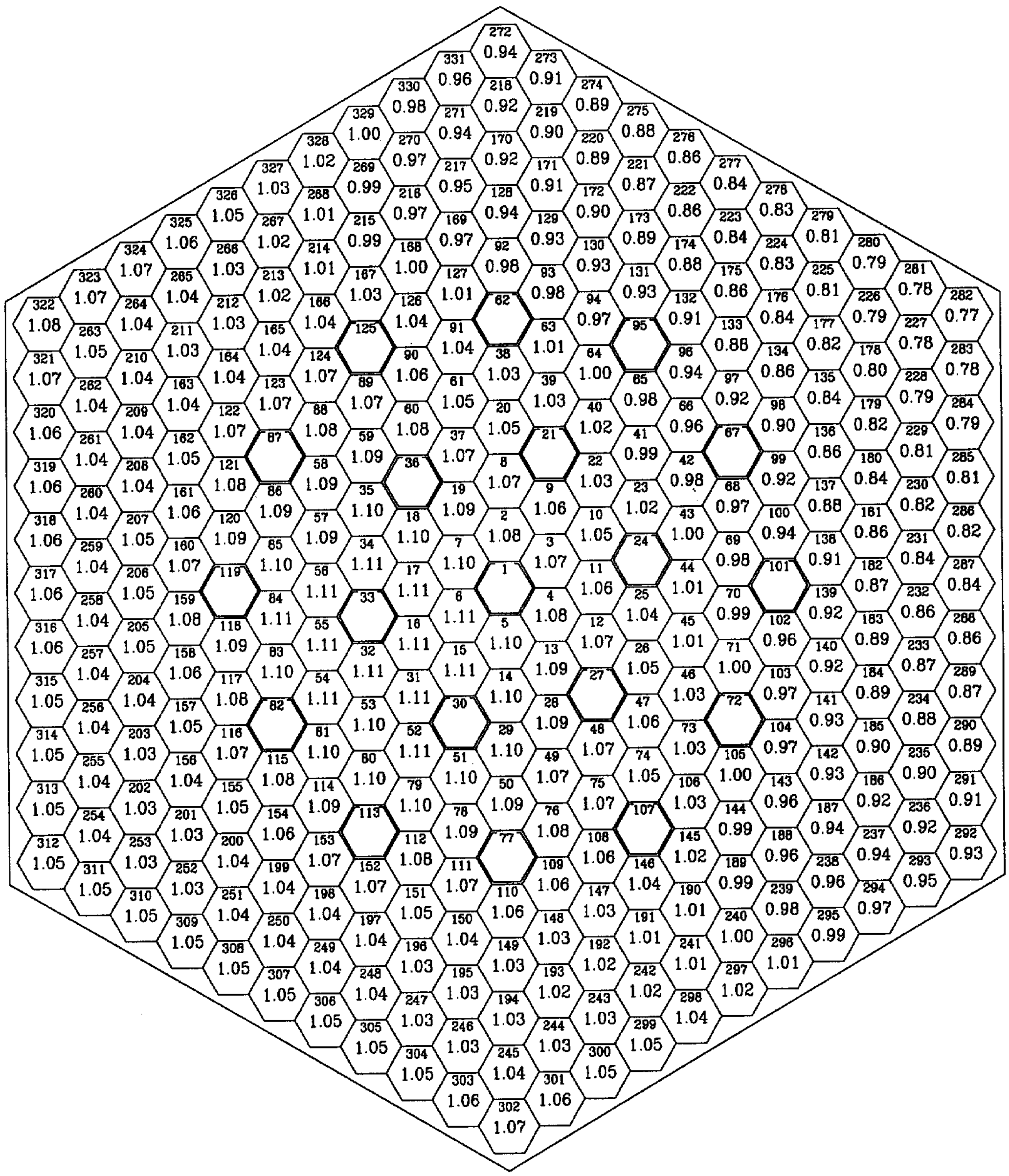
T	=	0.00 EFPD
W	=	3000.0 MW
$C_{H_3BO_3}$	=	5.79 g/kg
Ql_{max}	=	303.5 W/cm
Fuel ass.	=	126
Level	=	4
Fuel rod	=	16

Fig.3.32. Assembly-by-Assembly Maximum Linear Power Distribution in EOC. Third Cycle with 3 MOX LTAs of «Island-2» Type (Pu3.8-2.8-U3.7)



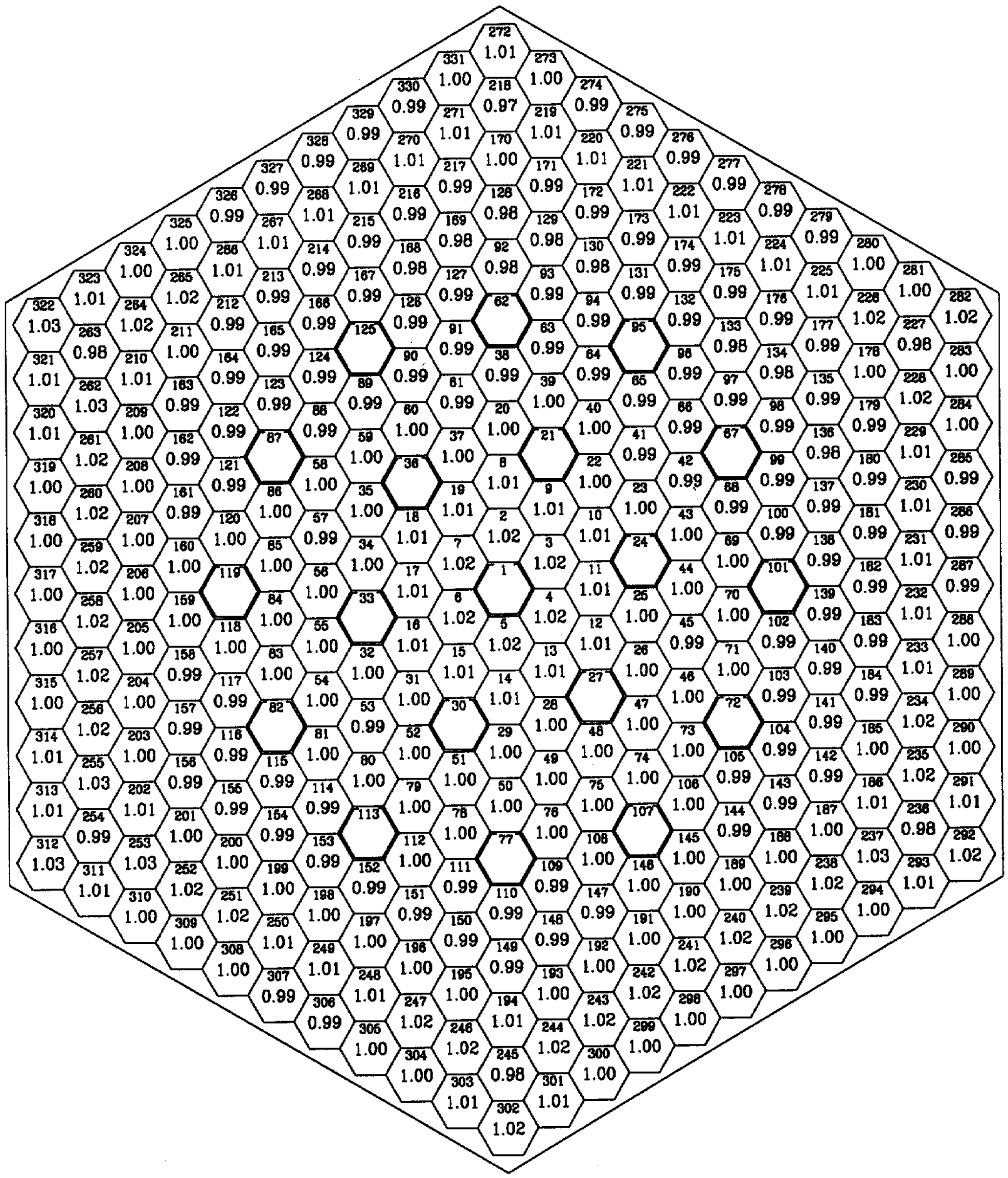
T	=	291.18 EFPD
W	=	3000.0 MW
$C_{H_3BO_3}$	=	0.00 g/kg
Ql_{max}	=	245.4 W/cm
Fuel ass.	=	124
Level	=	2
Fuel rod	=	255

Fig.3.33. Pin-by-Pin Power Distribution in the Most Powered Assembly in BOC. Third Cycle with 3 MOX LTAs of «Island-2» Type (Pu3.8-2.8-U3.7)



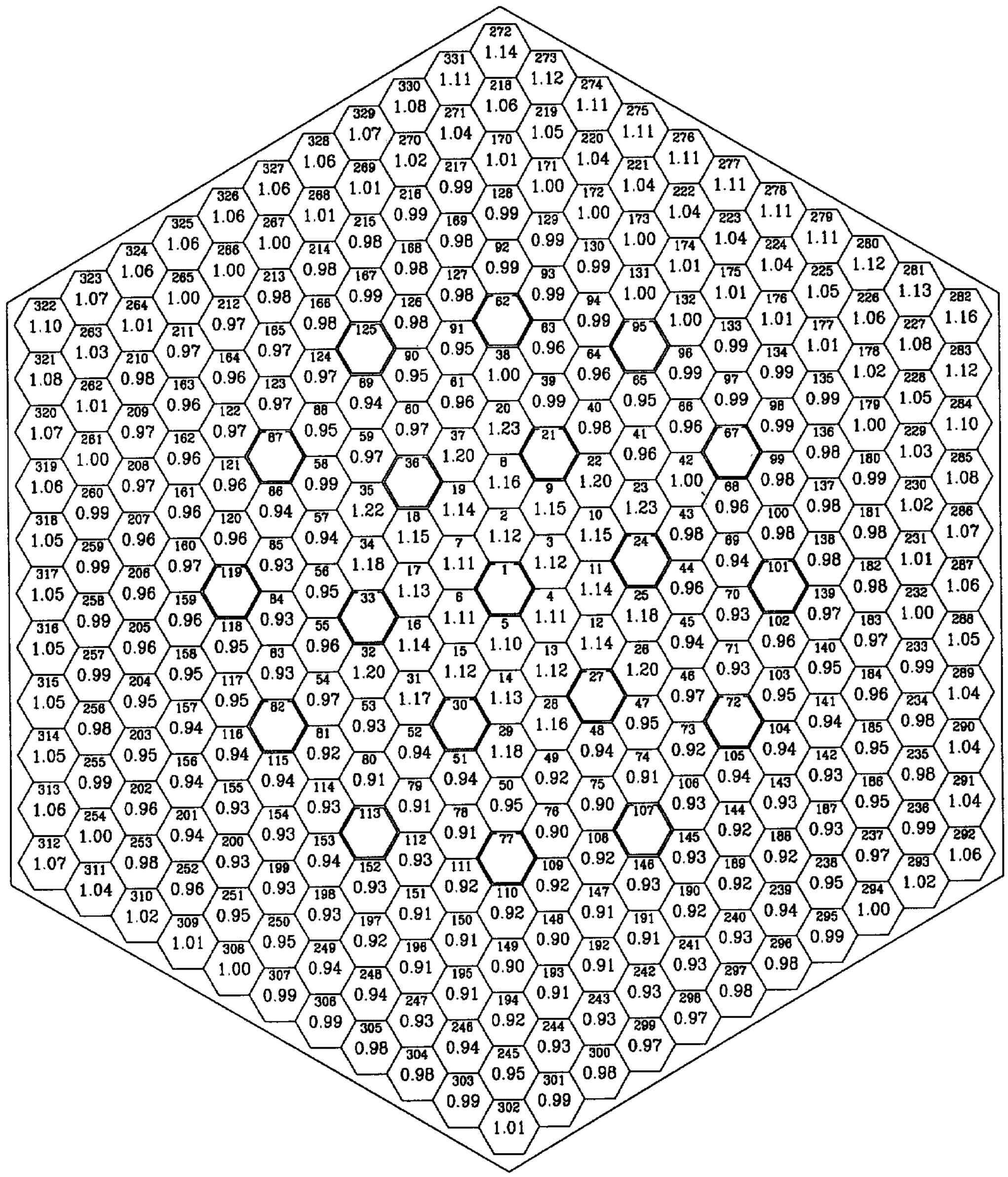
T	0.00	EFPD
W	3000.0	MW
$C_{H_3BO_3}$	5.79	g/kg
Ql_{max}	303.5	W/cm
Fuel assembly	126	
Level	4	
Fuel rod	16	
Kk_{max}	1.11	

Fig.3.34. Pin-by-Pin Power Distribution in the Most Powered Assembly in EOC. Third Cycle with 3 MOX LTAs of «Island-2» Type (Pu3.8-2.8-U3.7)



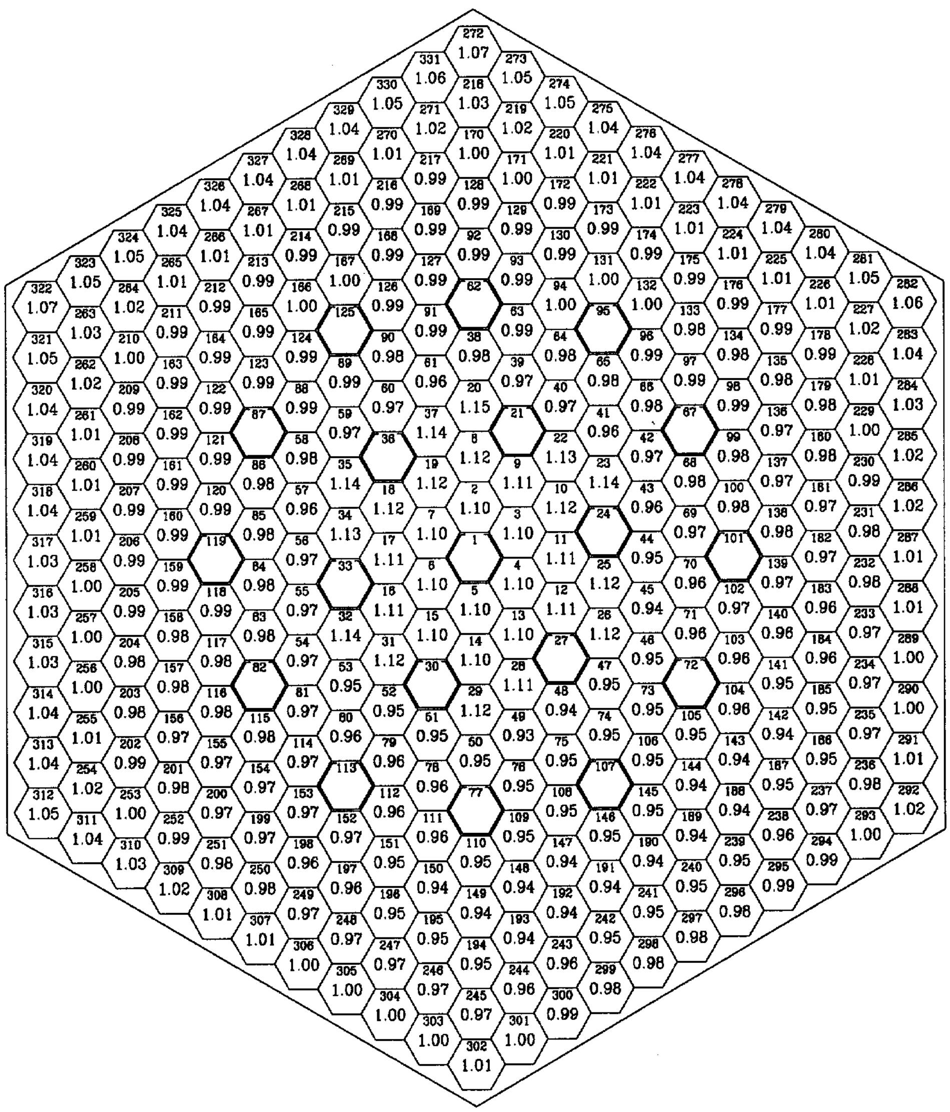
T	291.18	EFPD
W	3000.0	MW
$C_{H_3BO_3}$	0.00	g/kg
Ql_{max}	245.4	W/cm
Fuel assembly	124	
Level	2	
Fuel rod	255	
Kk_{max}	1.03	

Fig.3.35. Pin-by-Pin Power Distribution in MOX LTA in BOC. Third Cycle with 3 MOX LTAs of «Island-2» Type (Pu3.8-2.8-U3.7)



T	0.00	EFPD
W	3000.0	MW
$C_{H_3BO_3}$	5.79	g/kg
Ql	249.1	W/cm
Fuel assembly	111	
Level	4	
Fuel rod	20	
Kk_{max}	1.23	

Fig.3.36. Pin-by-Pin Power Distribution in MOX LTA in EOC. Third Cycle with 3 MOX LTAs of «Island-2» Type (Pu3.8-2.8-U3.7)



T	291.18	EFPD
W	3000.0	MW
$C_{H_3BO_3}$	0.00	g/kg
Ql	210.8	W/cm
Fuel assembly	111	
Level	4	
Fuel rod	20	
Kk_{max}	1.15	

ANNEX

A.1. Cell Code TVS-M

Nuclear data libraries

The nuclear data library is based on the same files of estimated nuclear data as precision code MCU-RFFI [1*], which uses the Monte Carlo method.

In the epithermal energy region (E>0.625 eV) the calculation is based on slightly modified microcross section library BNAB (see, e.g., [2]) with 24 energy groups. The nuclide libraries can contain both the group and subgroup constants and for some nuclides with temperature dependence.

For the calculation of neutron spectrum in the energy region of resolved resonances E_n<1 keV (15 and higher BNAB group) the library includes files of resonance parameters of individual nuclides obtained on the base of the LIPAR library. For all fissile nuclei the library contains prompt and delayed neutron spectra, group β values and decay constants for six groups of delayed neutrons.

The thermal energy region is divided into 24 groups. For the nuclides with the "l/v" cross-section behavior the absorption cross sections at 2200 m/s are used, for the rest ones the group values of the absorption, scattering and fission cross sections are specified. In addition, for oxygen and carbon the scattering matrices obtained in terms of gas model at 300, 373, 473, 558, 623K are given. For hydrogen bonded in water molecule the scattering matrix is obtained from the ENDF/B recommended data in terms of the Koppel model [3] at the same temperatures.

The library contains the files of cross sections and yields of 98 fission products including ^{135}Xe and ^{149}Sm. The files of fission product yields are based on the ENDF/B-VI data [4].

Uniform lattice

In the energy region of epithermal neutrons (10.5MeV<E_n<0.625 eV, BNAB groups 1-24) a detailed calculation of group spatial-energy distribution of neutron flux is performed. Each group is divided into an arbitrary number of intervals equal in lethargy, and then the calculation is performed at each point of group division. The of elastic scattering process is calculated without use of any approximations when the scattering is isotropic in the inertia center system (i.e.s), otherwise the scattering anisotropy is taken into account by the term not higher than linear in cosine of scattering angle. The slowing down due to inelastic scattering is taken into account via the matrix of inelastic transitions under the assumption of uniform energy distribution of neutrons scattering into the given group.

For nuclides with the subgroup description of cross sections the heterogeneous subgroup calculation of their micro cross sections is performed.

In the energy region of resolved resonances (groups 13-24 BNAB) for resonance nuclides the calculation of all types of cross sections is performed with the use of nuclide

* References in p.A.1 are placed in the end of A.1

resonance parameters. In so doing it is possible to take into account temperature dependence of resonance cross sections.

In the thermal energy region the standard calculation technique is used. It suggests solving the multigroup equation of thermalization with the neutron sources from the epithermal energy region formed when calculation for this energy range was performed.

Calculation of neutron spatial distribution is carried out by dividing the cells into an arbitrary number of annular material zones and by the use of the passing through probability (PTP) method [5]. In the calculation the actual form of the cell boundary is taken into account.

The calculation of the point kinetics parameters β_{eff}, ℓ is made by the standard formulas using the value function ψ with respect to K_{eff} and with six groups of delayed neutrons.

The calculation of the fuel nuclide composition during fuel burnup is performed for heavy nuclides from ^{232}Th to ^{244}Cm and for 98 fission products from ^{82}Kr to ^{163}Dy. The burnup equations can be solved both by the Runge-Kutt method and by a faster analytical method described in [6].

Calculation of supercells and fuel assemblies

For the determination of FA neutronic characteristics the code uses the diffusion fine-mesh calculation with an arbitrary number of groups from 4 to 48 and with the mesh width equal to the pitch between fuel rods in the FA. For the boundary mesh cells the compression coefficient is used. Along with the standard six-point scheme the refined scheme whose principles of construction are described in [7] can be used. The mesh equation has a common form however the quantities in this formula have another sense, namely:

$$\frac{4}{3a^2}\sum_{i=1}^{6}\frac{d_0 d_i}{d_0+d_i}(F_0-F_i)+(\Lambda_0^a+\Lambda_0^r+G_0^Z B_z^2)F_0=S_0 \qquad (1)$$

$$F=\varepsilon\Phi \qquad \Lambda=\Sigma/\varepsilon$$
$$G^Z=D^Z/\varepsilon \qquad d=D^R\xi \qquad (2)$$
$$\varepsilon=\psi(1-\gamma/\delta) \qquad \delta=2d/a$$

In formulas (2-7) Φ is the cell neutron flux; the sense of quantities Σ, D^R, D^Z is obvious. Then

$$\psi=\frac{\Phi_b^s}{\bar{\Phi}^s} \qquad \xi=\frac{j_b^a}{\bar{j}^a} \qquad (8)$$

Here Φ is the neutron flux in the given mesh cell; j is the neutron current in the cell; index "*b*" means the value of corresponding quantity at the cell boundary; index "*s*" indicates the solution of transport equation in the cell with symmetric boundary conditions (symmetric inflowing and outflowing neutron current); index "*a*" is the solution with asymmetric boundary conditions (neutron current flowing through the cell); the bar shows the quantity value averaged over the cell.

The use of these quantities permits joining of *accurate* (i.e. obtained from solving of transport equation for the cell) neutron flux and current at the cell boundary and keeping of the *accurate* connection between the solution of equation (1) and the reaction rates in the cell. In this way it becomes possible to avoid errors peculiar to the standard calculation scheme associated with the finite size and heterogeneous structure of mesh points. For solving the set of equations any modules of diffusion equation solutions can be used.

As usual the process of solving the diffusion equations is divided into the solving of the equation for each group and the determination of fission source by means of external iterations. If the state of FA at power is considered then upon their completion the external iterations are added with the calculation of ^{135}Xe and ^{149}Sm concentrations and a new iteration cycle.

Each mesh point pertains to a definite type: fuel rod, cell with absorber rod, cell corresponding the gap between FAs, etc. The constants for the background type are always calculated in the asymptotic mode, i.e. as for the uniform fuel cell. The constants for non-fuel cells are calculated in the mode of supercell. For the non-background fuel cells including those with integrated burnable poison (named tvegs) the calculation can be performed both in the asymptotic and supercell modes. The homogenized background cell is always considered as the external zone of supercell.

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A.2. Coarse-Mesh Code BIPR-7A

BIPR-7A is a 3-dimensional hexagonal coarse-mesh code intended to calculate neutronics characteristics of VVER-type reactor core.

Calculational cell represents assembly transversal section in horizontal plane and usually one-tenth of core height in axial direction i.e. there are 1630 cells in VVER-1000 core. Neutronics parameters are homogeneous within a cell.

Radial, upper and lower reflectors are described by border conditions.

Calculation is performed in two energetic groups using the so-called modal presentation of group fluxes [8].

Cell constants, prepared by the code TBC-M [4], form a library and represent a number of polynomials that reflect the two-group neutronics cross sections dependence on moderator density, moderator temperature, fuel temperature, FP concentrations in fuel, boron acid concentration in coolant, Xe and Sm concentration in fuel.

BIPR-7A is a part of industrial super-code KASKAD that allows obtaining in convenient formats all the parameters necessary for reactor safety estimations and licensing.

As a result BIPR-7A calculate the following parameters:

- q_i,
- Kq,
- q_{ij},
- Kv,
- BUi,
- BUij,
- MTC,
- MDC,
- DTC,
- DRO/DCB,
- βeff,
- λm,
- Cb_{CRIT} ,
- RO_{STOP},
- $(RO)_{AP}$.

A.3. Fine-Mesh Code PERMAK-A

PERMAK-A is a 2-dimensional fine-mesh code intended to calculate neutronics characteristics of VVER-type reactor core.

Calculational cell represents fuel pin-type hexagonal cell with homogeneous neutronics parameters within it.

Diffusion finite-differencies neutron balance equation in few energetic groups are resolved.

Radial reflector is described by the same manner as a core.

Neutron flux axial gradients, obtained by BIPR-7A, are used while calculating one (as usual) the most powered core axial level.

Cell (fuel and non-fuel) constants, prepared by the code TBC-M [4], form a special library and represent a number of polynomials that reflect the group neutronics cross sections dependence on moderator density, moderator temperature, fuel temperature, FP concentrations in fuel, boron acid concentration in coolant, Xe and Sm concentration in fuel.

PERMAK-A is a part of industrial super-code KASKAD that allows obtaining in convenient formats all the parameters necessary for reactor safety estimations and licensing.

As a result PERMAK-A calculates the following parameters:

- q_k,
- Kk,
- Kr;
- BUk,
- Q_l,
- $K_{o\text{-}total}$.

A.4. Reflector Description

The simplified structure of VVER-1000 radial reflector is presented in Fig. A.2.. In KI fine-mesh calculations by the code PERMAK-A the radial VVER-1000 reflector is modeled by "reflector assemblies" of five types (Figures A.1, A.3-A.7). Zero flux is applied on the outer reflector borders. The corresponding geometric condensation factors are applied to the cell types of reflector if the cells are situated in "reflector assembly" corners or on the borders.

The upper and lower reflectors can be described on the base of reactor core design presented in [1].

Figure A.1. Equilibrium Loading Pattern for Base Uranium Core with Boron BPRs, Core 60 ° Sector

An - 3,7% Enrichment graded by 3,3% Replaced After 3 Years (No BPRs)

Bn - 3,7% Enrichment graded by 3,3% Replaced After 3/4 Years (0,065 g/cc Boron BPRs, removed after 1st year)

Cn - 3,3% Enrichment Replaced After 3 Years (No BPRs)

1-28 - Positions in 60 Degree Sector

n - Current Cycle of Fuel Assembly

Symbol e.g. **$** - Position of a Fuel Assembly During Lifetime in Core

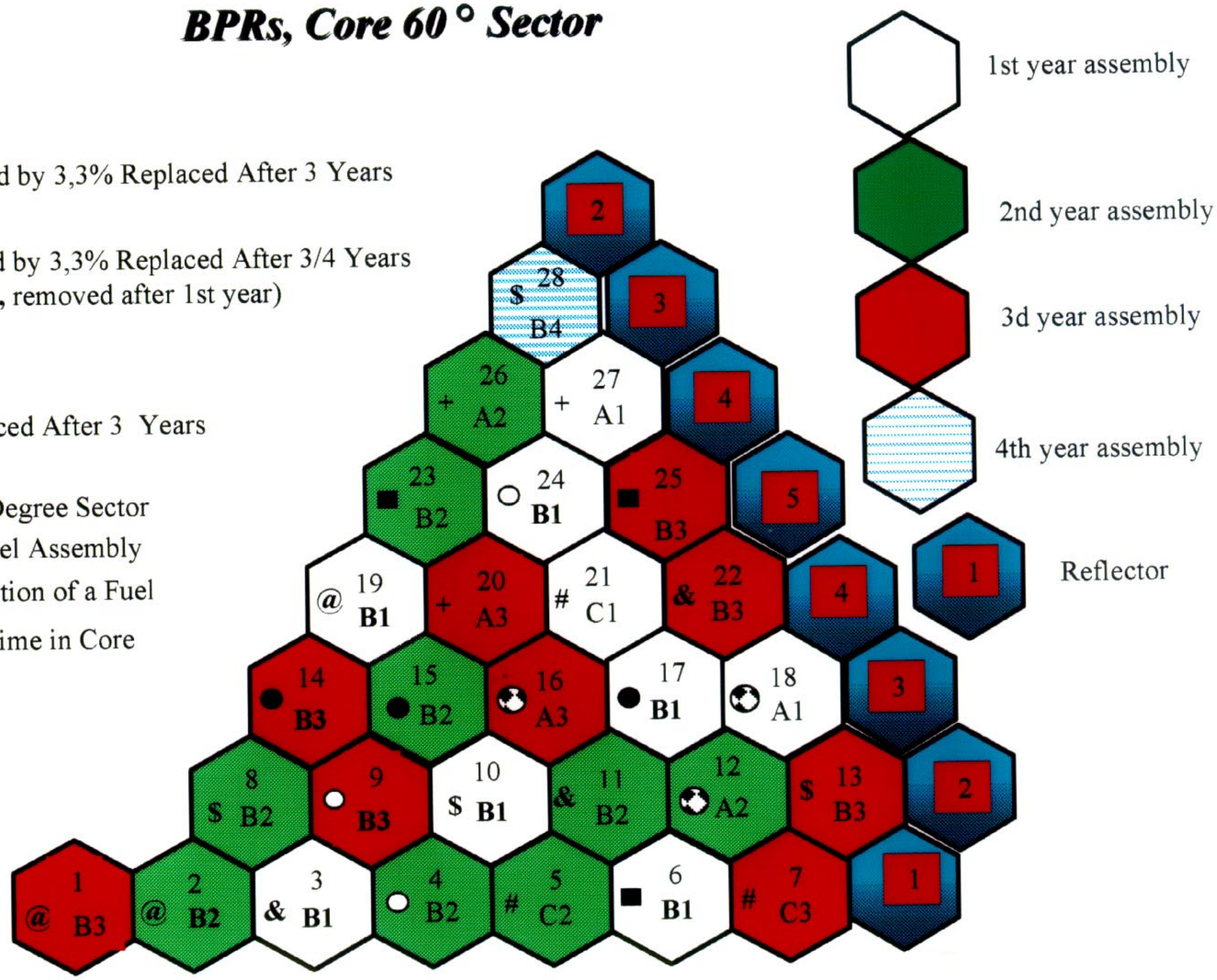
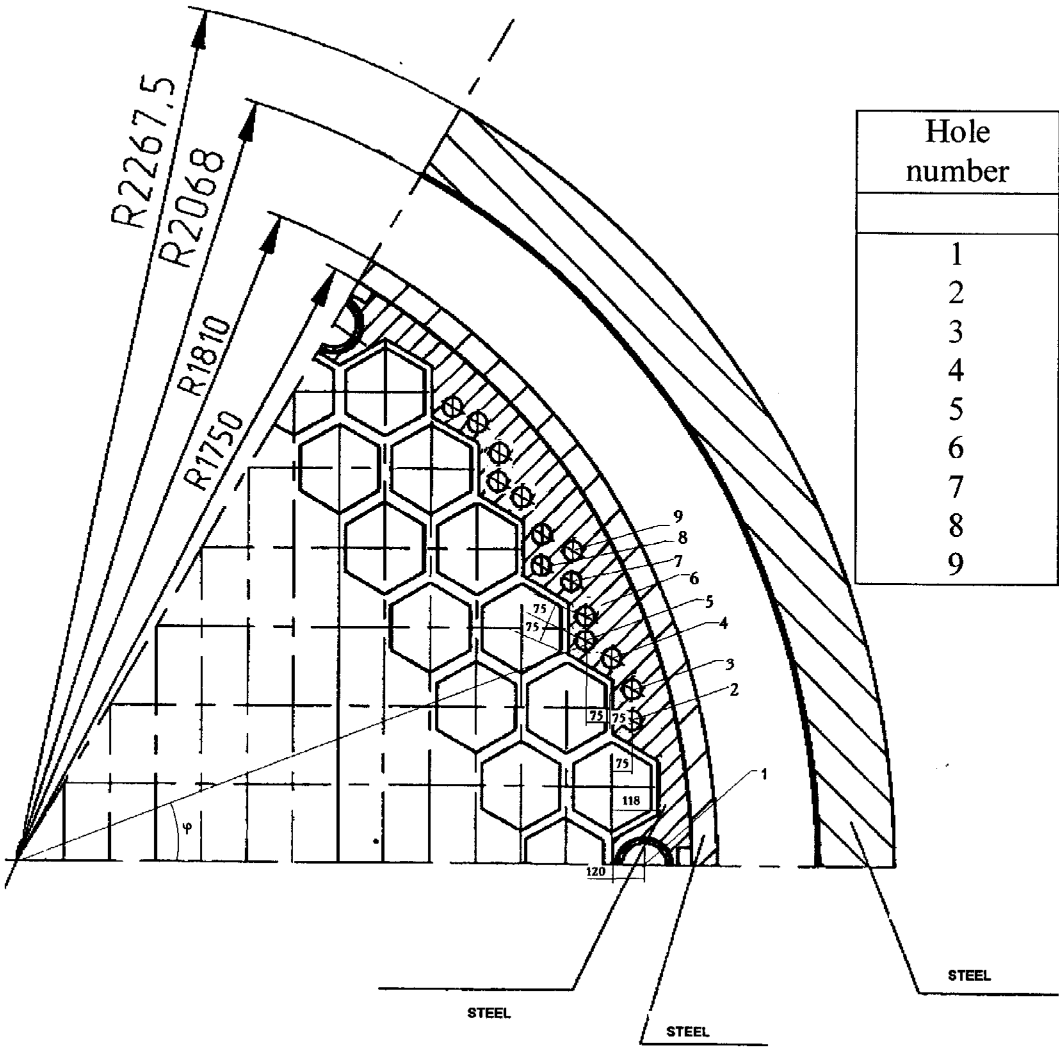


Figure A.2. Model of VVER-1000 Radial Reflector



Hole number	Distance from core center (R)	Angle (φ°)	Hole diameter
	mm		mm
1	1655	0	98
2	1655	13	70
3	1675	16	70
4	1655	19	70
5	1600	21	70
6	1635	24	70
7	1625	27	70
8	1575	30	70
9	1665	30	70

Fig.A.3. Reflector "assembly" of type 1

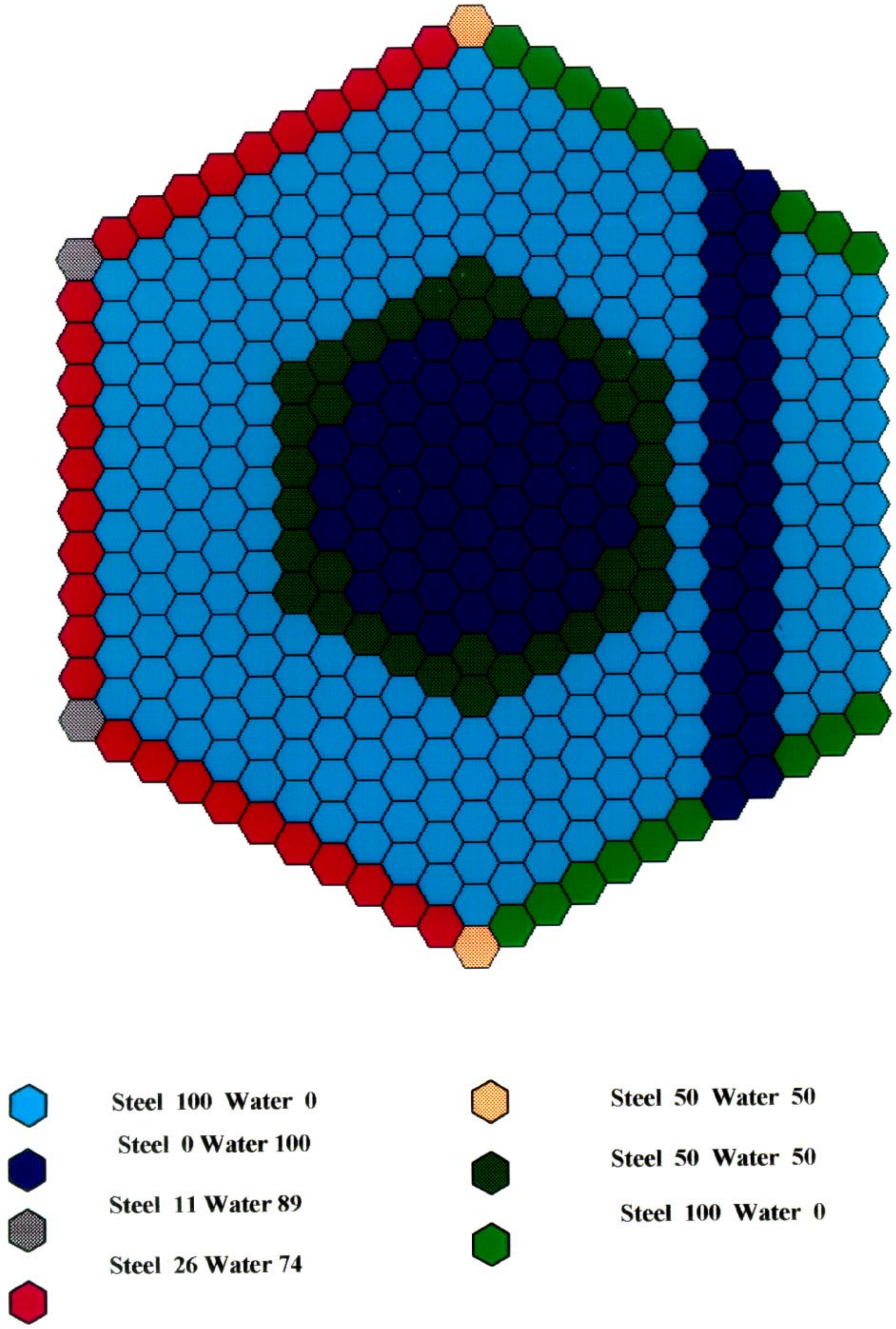


Fig.A.4. Reflector "assembly" of type 2

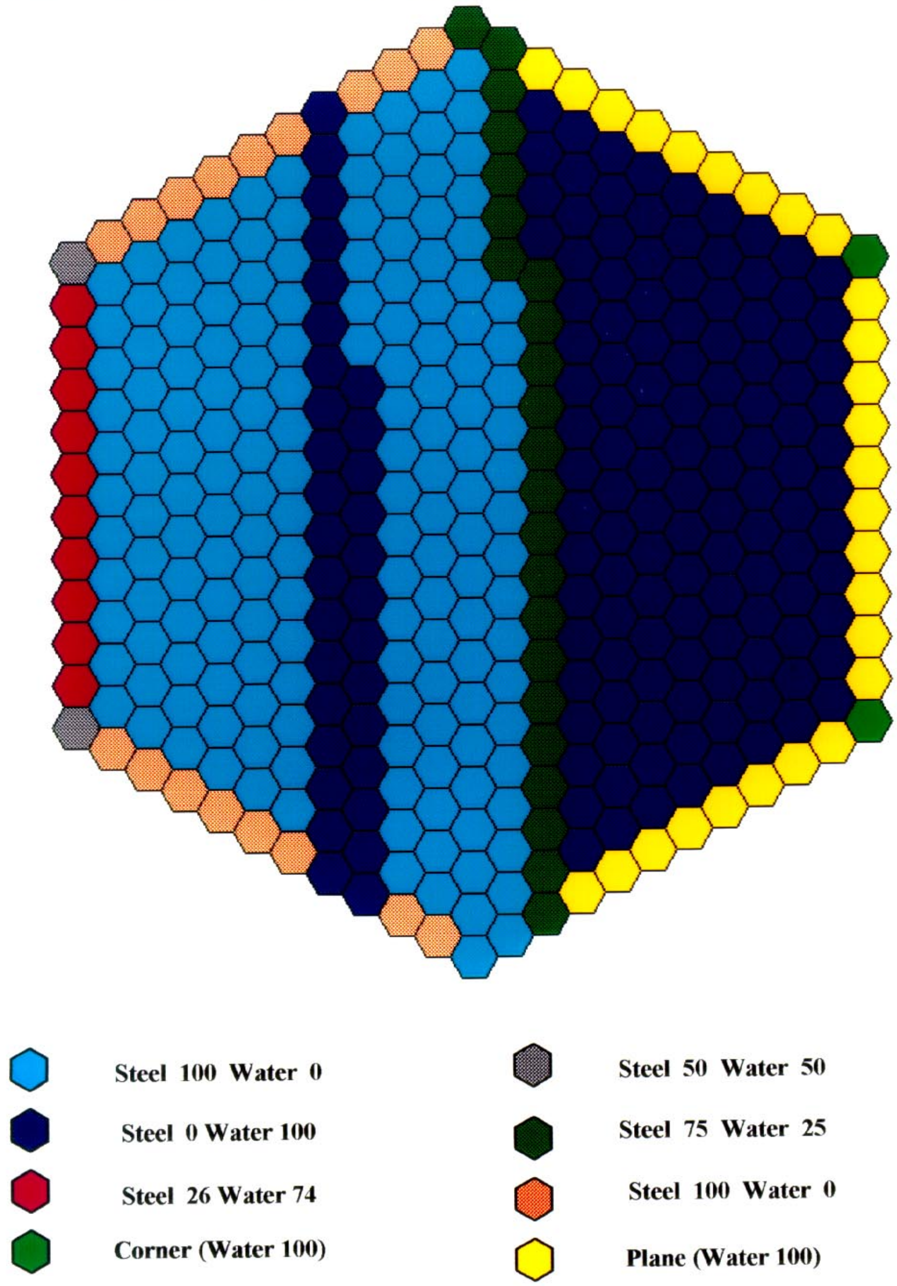
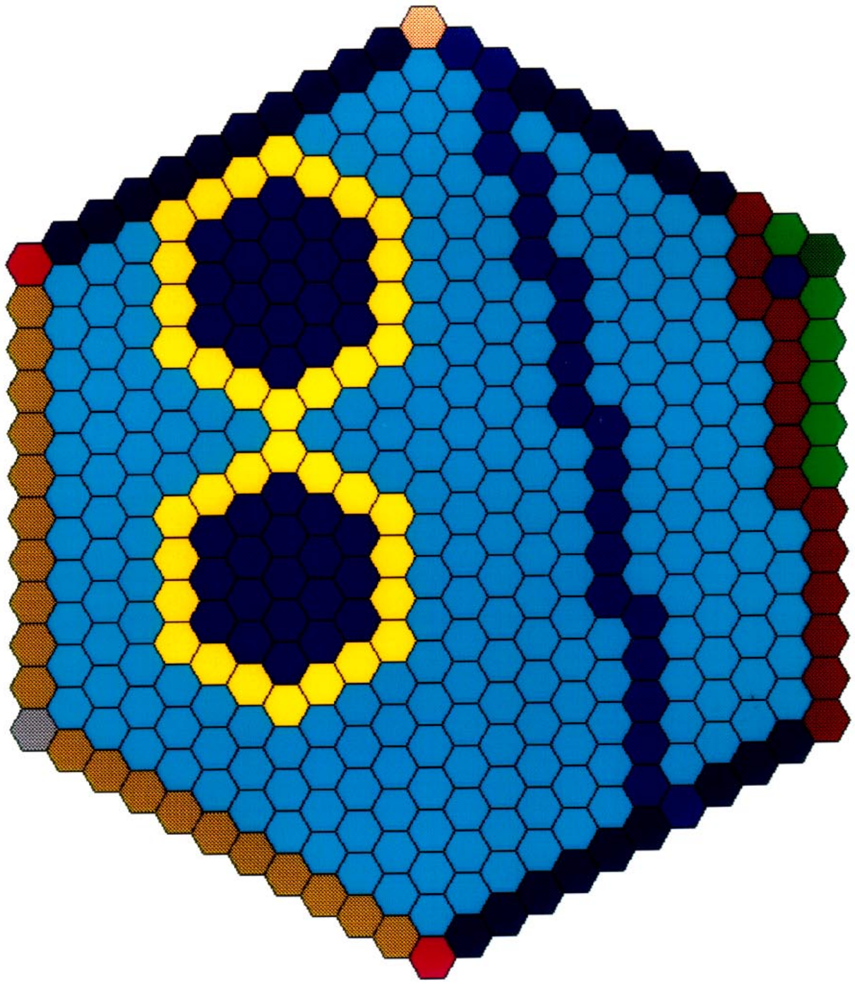


Fig.A.5. Reflector "assembly" of type 3



Steel 100 Water 0	Steel 11 Water 89
Steel 0 Water 100	Corner (Water 100)
Steel 50 Water 50	Steel 100 Water 0
Plane (Water 100)	Steel 50 Water 50
Steel 75 Water 25	Steel 26 Water 74
Steel 100 Water 0	

Fig.A.6. Reflector "assembly" of type 4

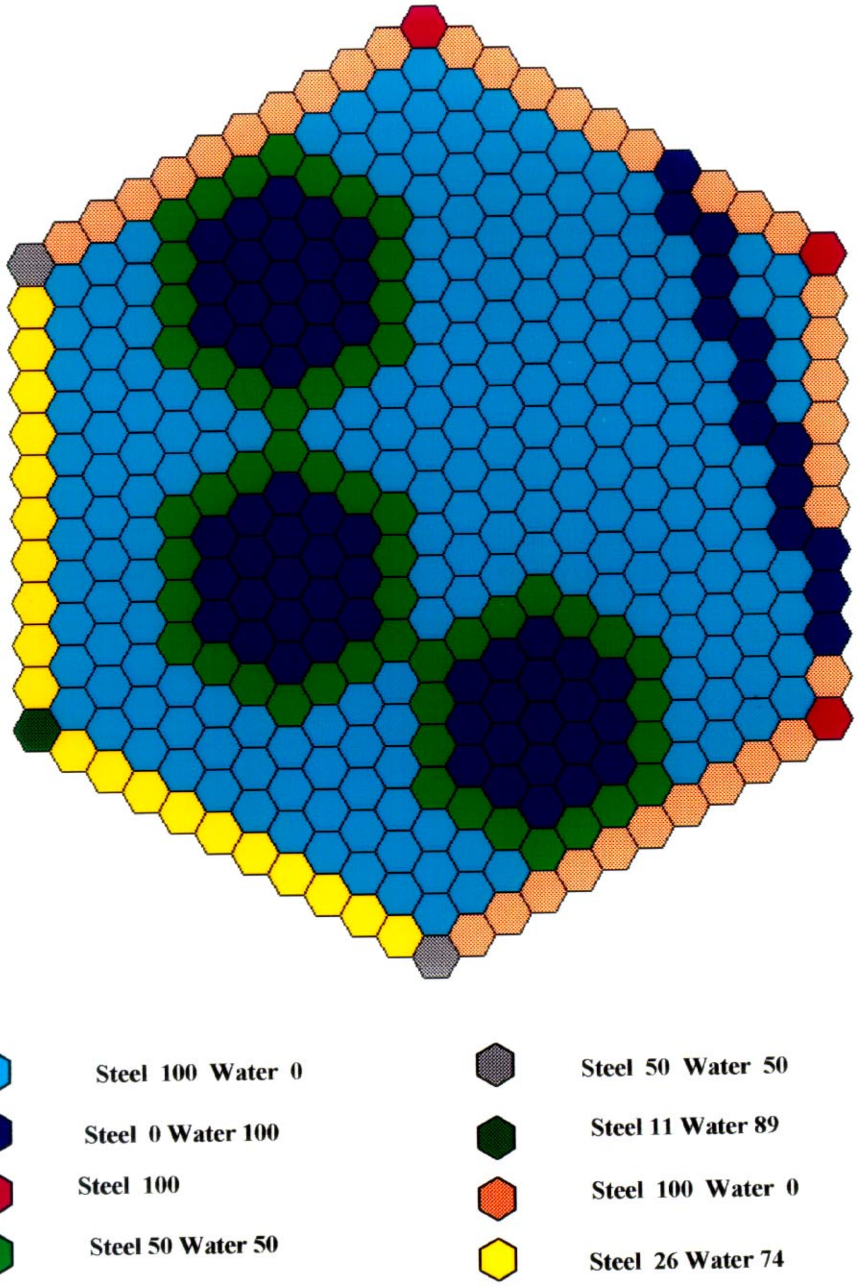
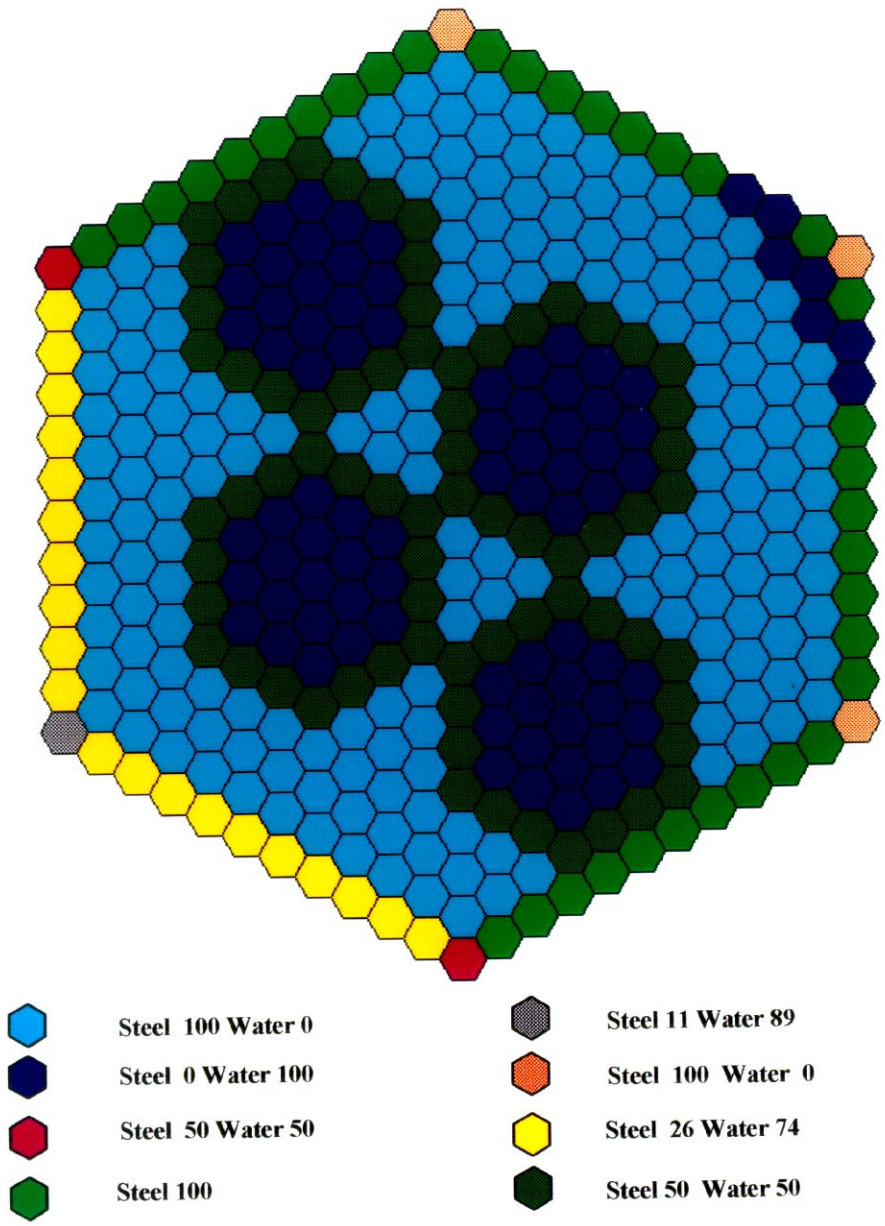


Fig.A.7. Reflector "assembly" of type 5



Comments from ORNL staff on the report, *Design Studies of “Island” Type MOX Lead Test Assembly*

1. Page 15. For the fifth row in the table, “2-D power peaking factor in assembly,” second column, the word “exploitation” is assumed to mean “burnup.”

2. Page 20. Currently the “island” option is not being pursued by the Fissile Materials Disposition Program. If, in the future, further studies are performed, depletion (burnup) calculations in US studies would be performed with a computational model in which the LTA is surrounded by uranium assemblies. Such a model will yield burnup-dependent data that is different (maybe not significantly) from a single-MOX-bundle model. However, Styrine reports that TVS-M models (infinite lattice of MOX LTAs) as reported in this report are properly adapted for BIPR calculations. Constants used in BIPR are supposed to be calculated with an asymptotic spectrum of an infinite grid. In RF studies, RF staff find an acceptable (from the point of view of power peaking values in core) plutonium grading in an infinite lattice of MOX LTAs. The parametric calculations reported here approach as close as possible to real situations in core management with BIPR. Plutonium grading is the only “initial data” that is passed to BIPR. Constants for BIPR are prepared by TVS-M for an infinite grid of fuel assemblies with the defined grading.

3. Page 21 and Table 2.9. It is noted that the burnable poison rods (BPR) in the uranium assembly are removed from the assembly after one cycle of irradiation, as is the case for U.S. reactors. While Table 2.9 shows only Ko evolution during irradiation for TVS-M calculation, really, of course, irradiation values more than ~16 MWd/kg for FA with Boron BPRs will not be reached.

4. Page 22 and Figures 2.41–2.43. The ratio F_1/F_2 and F_1 are spectral indices but the definitions of these indices are not provided. Styrine reports that F1 and F2 are, correspondingly, fast and thermal fluxes. Lazarenko reports that F1 is a neutron flux (in relative units) for the energy region from 0.625 eV to 10.5 MeV. It demonstrates the spatial distribution of fast and slowing down neutrons in assembly with the “island” configuration. The energy boundary between F1 and F2 is 0.625 eV. F2 is a thermal neutron flux for the energy region 0. to 0.625 eV. F1 and F2 have obtained from 48-group calculation by condensing procedure (F1—from 1–24 groups, F2—from 24–48 groups).

5. This report is the deliverable for FY 1999 Annual Operating Plan Task 10.2.2.1, milestone d. This milestone also had the internal ORNL designation of 99-1.

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