OAK RIDGE
NATIONAL LABORATORY

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FOR THE DEPARTMENT OF ENERGY

ORNL/SUB/00-85B99398V-13

Variant 22: Spatially-Dependent
Transient Processes in MOX Fuelled Core

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A Russian Contribution to the
Fissile Materials Disposition Program

UT-BATTELLE

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LTA PHYSICS DESIGN: DESCRIPTION OF ALL MOX PIN LTA DESIGN

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

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

Variant 22. Spatially-Dependent Transient Processes in MOX Fuelled Core

General Order 85B-99398V
(Report)

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Moscow 2000

ACRONYMS

Russian		West Equivalent
AZ	emergency (accident) protection	AP
AZ-1	state with all the contrȯl 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
BRU-A	Atmospheric Steam Dump (PG Relief Valves)	
DNB	Departure from Nucleate Boiling (passing of Critical Heat Flux)	DNB
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 Controllablė 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
PG	Steam Generator	SG
PWR	Pressurized-Water Reactor	PWR
RCT	Repeat Criticality Temperature	RCT
SUZ	Reactor Control and Protection System	RPS
TVS, FA	Fuel Assembly	FA
UOX, LEU	Uranium Oxide Fuel (Light Enrichment Uranium)	UOX, LEU
VVER	Russian water-water reactor	VVER

Executive Summary

In this document issued according to **Work Release 02. P. 99-4a** the reactor-kinetics benchmarks are presented for 30% MOX fuelled core of VVER-1000. Both UOX and MOX cores are calculated for three types of accidents:

- control rod ejection,
- overcooling of the reactor core caused by steam line rupture,
- boron dilution of coolant.

CONTENTS

INTRODUCTION

This work is a part of Joint U.S. / Russian Project with Weapons-Grade Plutonium Disposition in VVER Reactors and presents the results of spatial kinetics calculational benchmarks.

The examinations were carried out with the following purposes:

- to verify one of spatial neutronic kinetics model elaborated in KI,
- to understand sensibility of the model to neutronics difference of UOX and MOX cores,
- to compare in future point and spatial kinetics models (on the base of a set of selected accidents) in view of eventual creation of RELAP option with 3D kinetics.

The document contains input data and results of model operation of three emergency dynamic processes in the VVER-1000 core:

- Central control rod ejection by pressure drop caused by destroying of the moving mechanism cover.
- Overcooling of the reactor core caused by steam line rupture and non-closure of steam generator stop valve.
- The boron dilution of coolant in part of the VVER-1000 core caused by penetration of the distillate slug into the core at start up of non-working loop.

These accidents have been applied to

- Uranium reference core that is the so-called Advanced VVER-1000 core with Zirconium fuel pins claddings and guide tubes. A number of assemblies contained 18 boron BPRs while first year operating.
- MOX core with about 30% MOX fuel.

At a solving it was supposed that MOX-fuel thermophysical characteristics are identical to uranium fuel ones.

The calculations were carried out with the help of the program NOSTRA / 1 /, simulating VVER dynamics that is briefly described in Chapter1.

Chapter 3 contains the description of reference Uranium and MOX cores that are used in calculations. The neutronics calculations of MOX core with about 30% MOX fuel are named "Variant 21".

Chapters 4-6 contain the calculational results of three above mentioned benchmark accidents that compose in a whole the "Variant 22".

1. Brief Description of Code NOSTRA

Code NOSTRA have been developed for safety parameters investigations of nuclear power plants (NPP) with reactors of VVER type. Code is intended for computational analysis of transient and accidental processes that may include arising of core multiplying properties ingomogeneity, for example failures of steam generator operation due to non-complete mixing of coolant in pressure chamber, "shooting" of alone cluster etc.

Code NOSTRA allows to obtain the time-dependent three-dimensional neutron and temperature distributions in core. Code is capable to model the variations of NPP equipment operation regime for known time-dependencies of flow rate, pressure and temperature of coolant at the core inlet and for known strategy of control rods movement.

Code allows to compute the transient and accidental regimes caused by the following reasons:

- failures of main circulation pumps;
- variation in steam generator load;
- failures in feed water system;
- rupture of pipelines in the 2-nd circuit;
- rupture of pipelines in feed water system;
- rupture of pipelines in auxiliary system of the 1-st circuit;
- action of reactor power control system;
- violation in action of reactor power control system, e.g. self-movement of control rod, sticking of cluster during operation etc.;
- throwing out the cluster;
- any combination of these reasons.

Code is capable to perform the operational computations for NPP with reactors VVER-440, VVER-1000 and design computations for advanced NPPs.

Neutronics processes are simulated on the basis of certified code BIPR-7 [2] methodology that was generalized on non-stationary equations with 6 groups of delayed neutron predecessors and non zero life-time of prompt neutrons. Code BIPR-7 considers 3-dimensional 1.5-group diffusion model of neutron transport. Diffusion equations are solved by the 'coarse-mesh' method with coefficients obtained with using of nodal algorithm. Asymptotic and transient functions of each node are utilized as modes. Additional balances are formed at the boundaries of considered prisms. Algorithm takes into account the discontinuity of diffusion properties on boundaries of nodes. For solving of the algebraic equations the non-linear iterative procedure is used that permits to reduce the dimensionality of the problem.

Neutronics characteristics of nodes are approximated by polynomials depending on the sort of fuel, fuel burnup, concentrations of boron and xenon, temperature distribution and state of coolant. The applicability of neutron algorithms is restricted to the steam content of no more than 10%.

Code solves the burnup equations and permit to analyse transients in any time of reactor operation. Model of residual heat generation takes into account the fuel burnup and is based on US Standard ANS-5.1.

Fuel and coolant temperature, concentrations of delayed neutrons predecessors are defined as average values for considered prisms formed in core space discretization as in codes of BIPR type.

Thermophysical model of the fuel elements considers the cladding, gas-gap and three fuel layers. Equations of heat and mass transfer are solved by explicit-implicit methods. Choice of method depends on the value of time step. Hydraulic computation is carried out for each assembly allowing to consider the situations with increased hydraulic resistance of some assemblies.

Code NOSTRA is provided by special system for graphic and tabular presentation of output data. Particularly, there is a capability to print two- and three-dimensional histograms of temperature and neutron fields. Code NOSTRA is forming the files of three-dimensional neutron fields for each step of time. It is possible to perform the time scanning of reactor parameters as animated cartoon.

Calculational subroutines are written on FORTRAN-language, service subroutines - on PASCAL-language. Code NOSTRA is operated on computers of IBM PC type.

2. Definitions

Parameter	Abbreviation	Units	Remarks
Calculational system	CS		Multi-assembly 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 or GWd/t	Burnup in Vij.
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	$\overline{B}_U$	MWd/kg or GWd/t	
Average burnup in MOX assemblies	$\overline{B}_{MOX}$	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 of fuel pins in core	q_{ijk}		Power of axial volumes of fuel pins normalised 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} (notation Kk or Kk_{max} is also used)		Maximum relative power of fuel pins (maximum in q_k values)

[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.

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	l_m or l_{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, 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

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

3. Equilibrium Fuel Cycle with 30% MOX Loaded Core (Variant 21)

Every accident calculation is to be executed for uranium reference core and for MOX fuelled core. The equilibrium fuel cycle of the "Uranium base core" with boron BPRs is presented in Figures 3-5 and 3-6. The simplified design of uranium fuel assemblies used here is shown in Figures 3-2, 3-3 and 3-4.

The first MOX core has been formed by introduction of 18 MOX assemblies into the uranium core periphery. The simplified design of MOX fuel assemblies is shown in Fig. 3-1. MOX fuel isotopic composition is presented in Table 3-1. The fissile plutonium enrichments are chosen, according to preliminary estimations, to ensure a reasonable value of pin power peaking factor in assembly. Equilibrium fuel cycle pattern of MOX fuelled core with 54 Plutonium assemblies is shown in Fig. 3-7 and Fig. 3-8. It is named "MOX core".

Characteristics of fuel rods, fuel assemblies, absorbers and boron poison rods are presented in Tables 3-2, 3-3, 3-4, 3-5, 3-6 and 3-7. Control rods grouping in VVER-1000 is presented in Fig. 3-9. The bank 10 is a regulating one inserted constantly into the core in the process of fuel irradiation. Assemblies' numeration is shown in Fig. 3-10.

The simplified structure of VVER-1000 radial reflector is presented in Fig. 3-11. In fine-mesh calculations that are carried out in KI the radial VVER-1000 reflector is modelled by "reflector assemblies" of five types (Figures 3-5, 3-12, 3-13, 3-14, 3-15 and 3-16). Zero flux is applied on the outer reflector borders.

Core modelling in axial direction is illustrated in Fig.3-17.

Fuel irradiation simulation is executed under the following conditions:

- The bank 10 is inserted of 20% core height all over a fuel cycle; the rest of banks are extracted;
- EOC corresponds to Cb crit =0;
- Equilibrium Sm and Xe;
- W = Wnom;
- t_{entry} = 287°C.

Table 3-1. Composition of Weapons Grade Plutonium

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

Table 3-2. 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	3.54
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
Reactor Coolant Flow Rate	m^3/hr	84000
Bypass Coolant Flow	%	3
Pressure at Core Inlet	MPa	16.1
Core Inlet Temperature	°C	287

Table 3-3. 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 3-4. Uranium 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
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 3-5. 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 3-6. 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	2.869	2.896	2.945
Absorber Composition		Boron g / cc		
		0.020	0.036	0.065
B10	W%	0.1278	0.2279	0.4046
B11		0.5694	1.0153	1.8028
Al		93.5246	91.7424	88.5951
Fe		0.1952	0.1915	0.1850
Ni		1.9525	1.9153	1.8496
Cr		1.6780	2.9923	5.3133
Zr		1.9525	1.9153	1.8496

Compositions Weight percent:

*

Zr	Nb	Hf
98.97	1.0	0.03

Table 3-7. Control Rod Design Parameters

Parameter	Units	Value
Clad Inner Diameter	cm	0.700
Clad Thickness	cm	0.06
Clad Material		Stainless Steel*
Absorber Diameter	cm	0.700
Absorber Material		Natural B4C**
Absorber Density	g / cc	1.80

Compositions Weight percent:

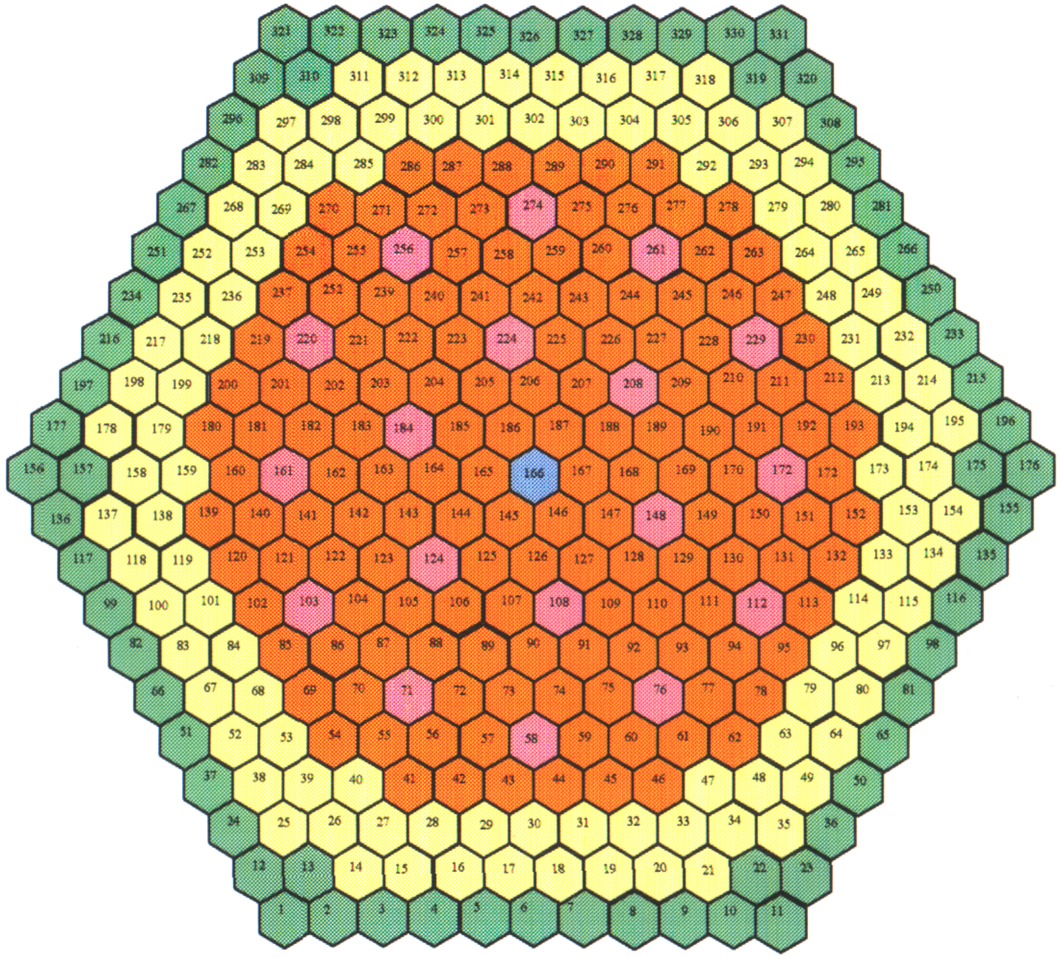
*

C	Cr	Ni	Ti	Fe
0.12	18.5	10.5	1.0	69.88

** **Content of ^{10}B is 19.8% atoms.**

Remark. The lower part (30 cm) of control rods consists of Dy_2O_3 TiO2 of density 4.9 g / cc.

Fig.3-1 . Simplified Design for 3-Zones MOX assembly



2.0% fissile Pu MOX Rods

3.0% fissile Pu MOX Rods

4.2% fissile Pu MOX Rods

Central tube

Guide Tube

Fig.3-2. Simplified Design for VVER-1000 Fuel Assembly. Type A

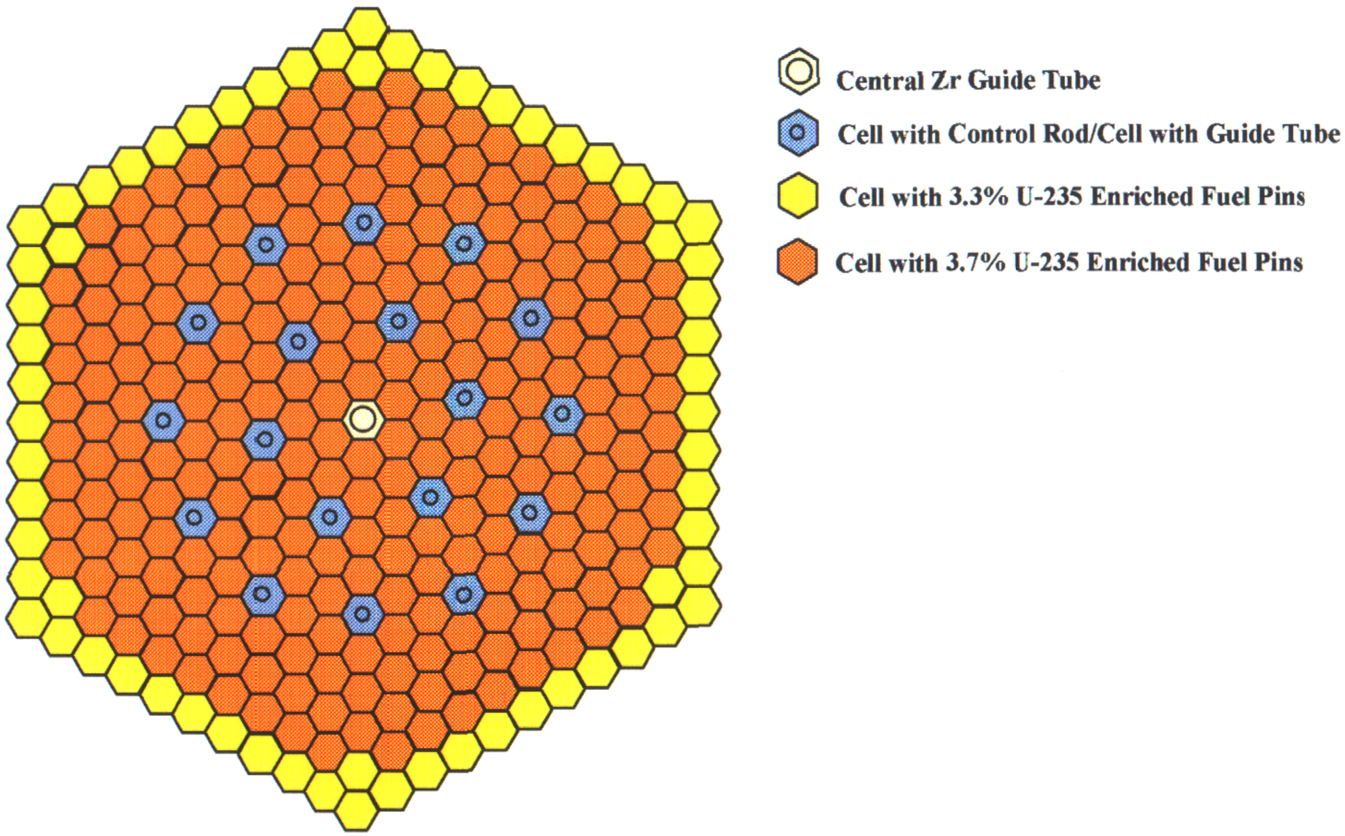
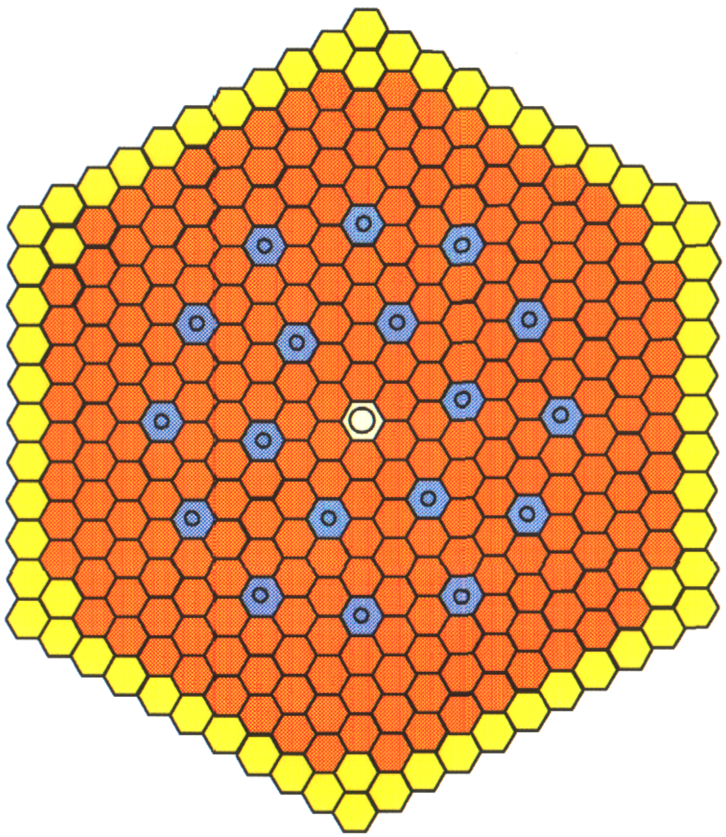


Fig.3-3. Simplified Design for VVER-1000 Fuel Assembly. Type B, Ba and Bb



Central Zr Guide Tube

Cell with Burnable Absorber/Cell with Guide Tube

Cell with 3.3% U-235 Enriched Fuel Pins

Cell with 3.7% U-235 Enriched Fuel Pins

B - (0,065 g/cc Boron BPRs, removed after 1st year)

Ba - (0,036 g/cc Boron BPRs, removed after 1st year)

Bb - (0,020 g/cc Boron BPRs, removed after 1st year)

Fig.3-4. Simplified Design for VVER-1000 Fuel Assembly. Type C

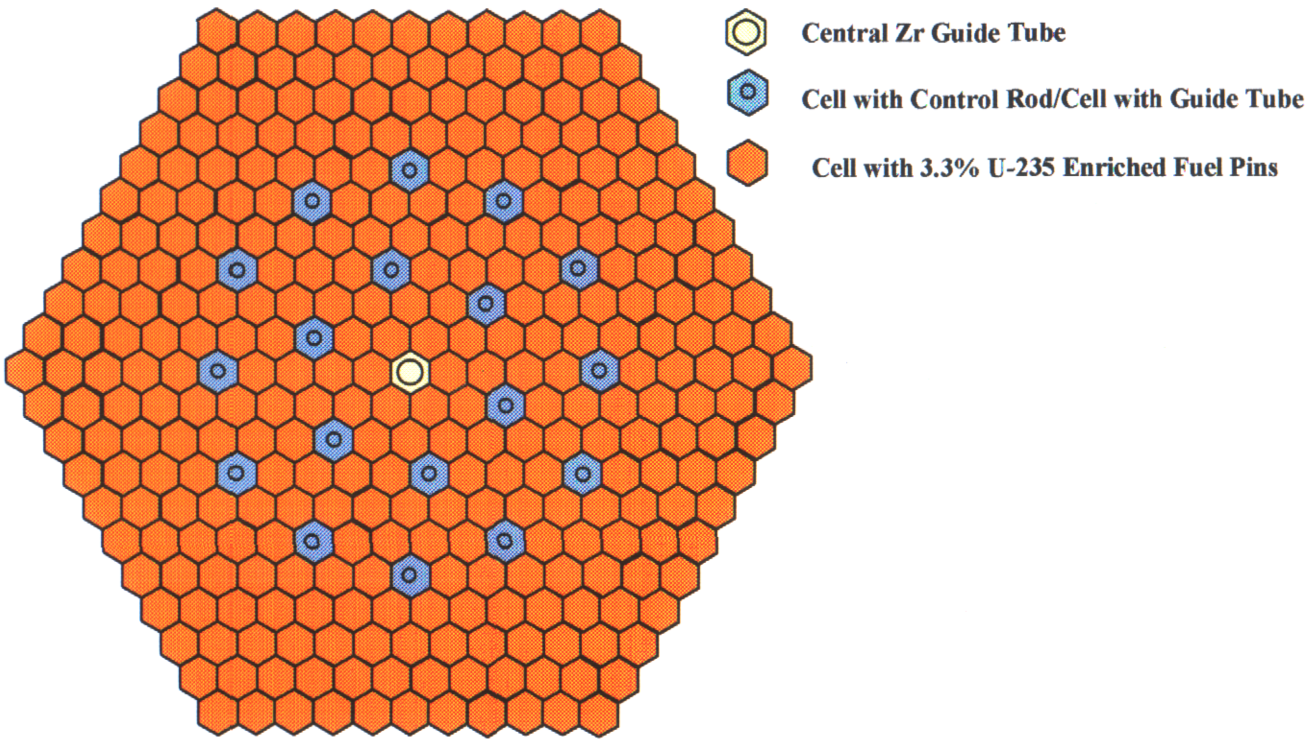


Fig.3-5. Equilibrium Loading Pattern for Base Uranium Core with Boron BPRs, Core 60° Sector

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

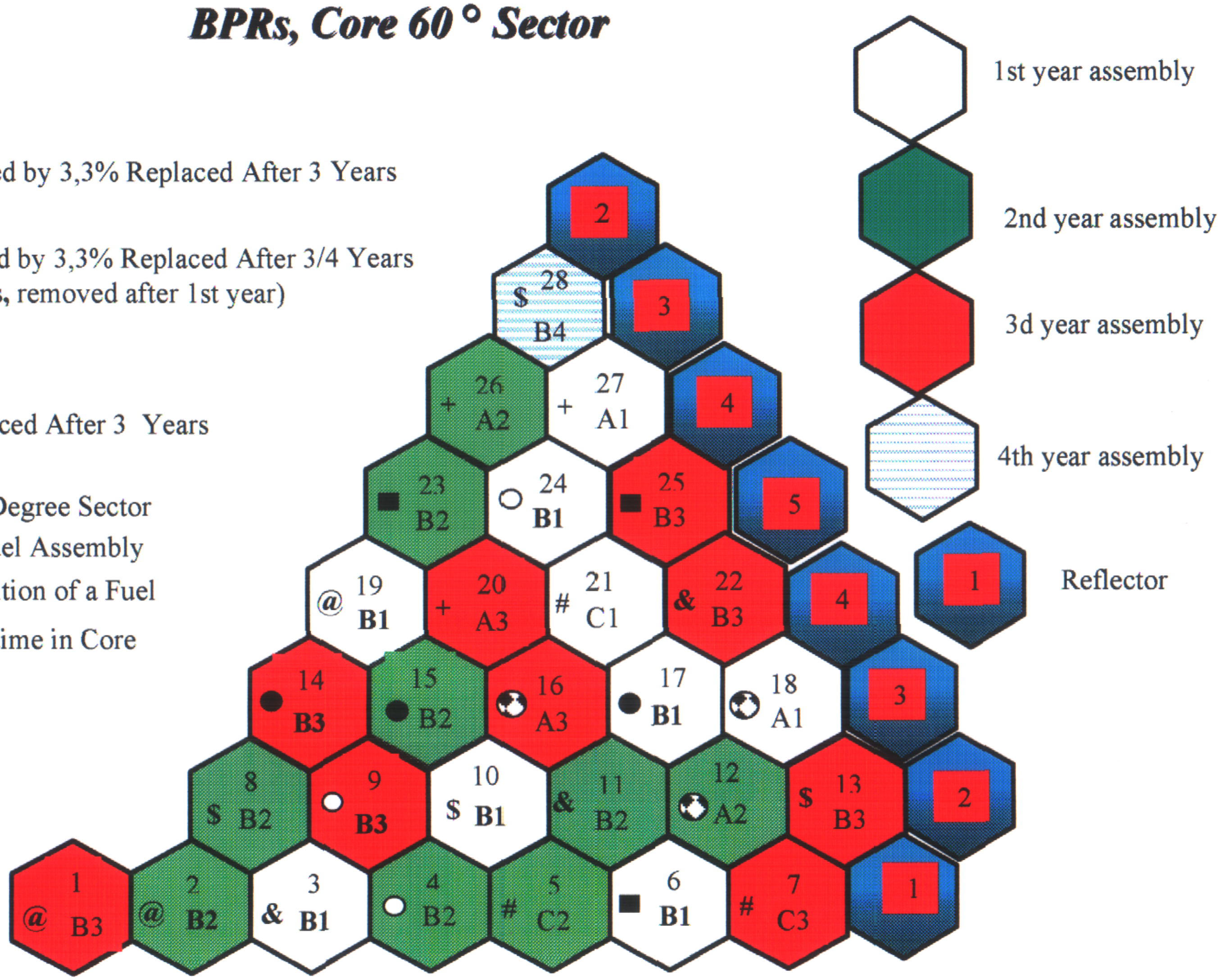
Bn - 3,7% Enrichment graded by 3,3% Replaced After 3/4 Years (0,065 g/cc Boron BPRs, removed after 1st year)

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

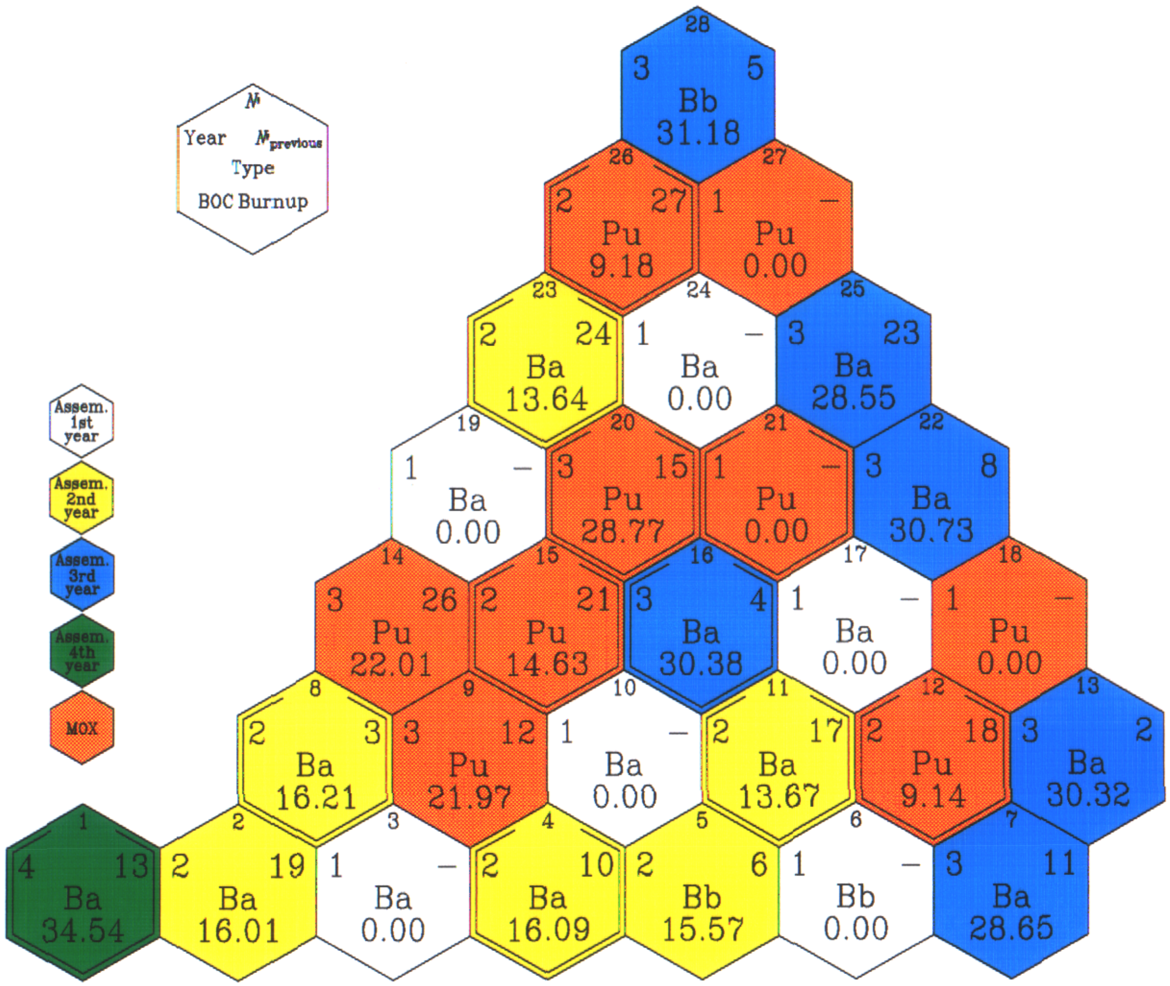


BURNUP(KG F-P/T FUEL) BOC
BURNUP(KG F-P/T FUEL) EOC
RELATIVE POWER BOC
RELATIVE POWER EOC

28						
3.4552E+01						
3.7970E+01						
2.9703E-01						
3.7042E-01						
26	27					
9.4878E+00	0.0000E+00					
2.0191E+01	7.9323E+00					
1.0155E+00	7.2646E-01					
1.0601E+00	8.0797E-01					
23	24	25				
1.4940E+01	0.0000E+00	3.0004E+01				
2.7518E+01	1.1910E+01	3.5474E+01				
1.2434E+00	1.1149E+00	4.9584E-01				
1.1993E+00	1.1848E+00	5.6691E-01				
19	20	21	22			
0.0000E+00	2.2363E+01	0.0000E+00	3.0316E+01			
1.3348E+01	3.3883E+01	1.2624E+01	3.5767E+01			
1.3011E+00	1.1420E+00	1.2370E+00	4.9398E-01			
1.2859E+00	1.0971E+00	1.2025E+00	5.6514E-01			
14	15	16	17	18		
2.9106E+01	1.4296E+01	2.2474E+01	0.0000E+00	0.0000E+00		
3.9180E+01	2.6576E+01	3.3950E+01	1.1882E+01	7.9817E+00		
1.0037E+00	1.2318E+00	1.1365E+00	1.1111E+00	7.3092E-01		
9.6843E-01	1.1608E+00	1.0941E+00	1.1831E+00	8.1290E-01		
8	9	10	11	12	13	
1.6166E+01	2.8873E+01	0.0000E+00	1.5729E+01	9.5718E+00	3.0550E+01	
2.8153E+01	3.8955E+01	1.3314E+01	2.8146E+01	2.0285E+01	3.4214E+01	
1.2283E+00	1.0024E+00	1.2964E+00	1.2247E+00	1.0154E+00	3.1949E-01	
1.1225E+00	9.7048E-01	1.2839E+00	1.1868E+00	1.0619E+00	3.9543E-01	
1	2	3	4	5	6	7
3.0565E+01	1.6180E+01	0.0000E+00	1.4294E+01	1.5026E+01	0.0000E+00	2.9914E+01
4.0498E+01	2.8270E+01	1.2886E+01	2.5894E+01	2.7341E+01	1.2419E+01	3.5531E+01
1.0298E+00	1.2611E+00	1.2748E+00	1.1859E+00	1.2374E+00	1.1684E+00	4.9960E-01
9.2825E-01	1.1160E+00	1.2372E+00	1.0874E+00	1.1626E+00	1.2325E+00	5.9583E-01

Fig. 3-6. Uranium Zone. Assembly Burnup, Relative Power

Fig.3-7. Equilibrium Loading Pattern for MOX Fueled Core with 54 MOX Fuel Assemblies, Core 60 ° Sector



BURNUP(KG F-P/T FUEL) BOC
BURNUP(KG F-P/T FUEL) EOC
RELATIVE POWER BOC
RELATIVE POWER EOC

28						
3.1184E+01						
3.4611E+01						
3.0911E-01						
3.6668E-01						
26	27					
9.1836E+00	0.0000E+00					
1.9963E+01	7.6534E+00					
1.0584E+00	7.5555E-01					
1.0438E+00	7.5180E-01					
23	24	25				
1.3641E+01	0.0000E+00	2.8555E+01				
2.6270E+01	1.1354E+01	3.3750E+01				
1.2428E+00	1.0802E+00	4.8501E-01				
1.2081E+00	1.1276E+00	5.4117E-01				
19	20	21	22			
0.0000E+00	2.8772E+01	0.0000E+00	3.0726E+01			
1.3391E+01	3.8934E+01	1.2318E+01	3.5754E+01			
1.2865E+00	9.8680E-01	1.2676E+00	4.6736E-01			
1.2901E+00	9.9787E-01	1.1672E+00	5.2600E-01			
14	15	16	17	18		
2.2005E+01	1.4628E+01	3.0380E+01	0.0000E+00	0.0000E+00		
3.3415E+01	2.6481E+01	3.9966E+01	1.1393E+01	7.6270E+00		
1.1244E+00	1.1721E+00	9.1902E-01	1.0780E+00	7.5038E-01		
1.0915E+00	1.1323E+00	9.5212E-01	1.1362E+00	7.5113E-01		
8	9	10	11	12	13	
1.6211E+01	2.1966E+01	0.0000E+00	1.3671E+01	9.1446E+00	3.0319E+01	
2.8526E+01	3.3381E+01	1.3474E+01	2.6372E+01	1.9925E+01	3.3796E+01	
1.2107E+00	1.1246E+00	1.2905E+00	1.2451E+00	1.0560E+00	3.1349E-01	
1.1759E+00	1.0923E+00	1.3013E+00	1.2189E+00	1.0458E+00	3.7203E-01	
1	2	3	4	5	6	7
3.4539E+01	1.6012E+01	0.0000E+00	1.6089E+01	1.5566E+01	0.0000E+00	2.8649E+01
4.3881E+01	2.8247E+01	1.3614E+01	2.7752E+01	2.8780E+01	1.3136E+01	3.4671E+01
8.9037E-01	1.1985E+00	1.3071E+00	1.1637E+00	1.3092E+00	1.2636E+00	5.5245E-01
9.2900E-01	1.1750E+00	1.3088E+00	1.1019E+00	1.2500E+00	1.2618E+00	6.2459E-01

Fig. 3-8. MOX Zone. Assembly Burnup, Relative Power.

Fig.3-9. Control Rods Grouping and Positions of in-core self-powered detectors

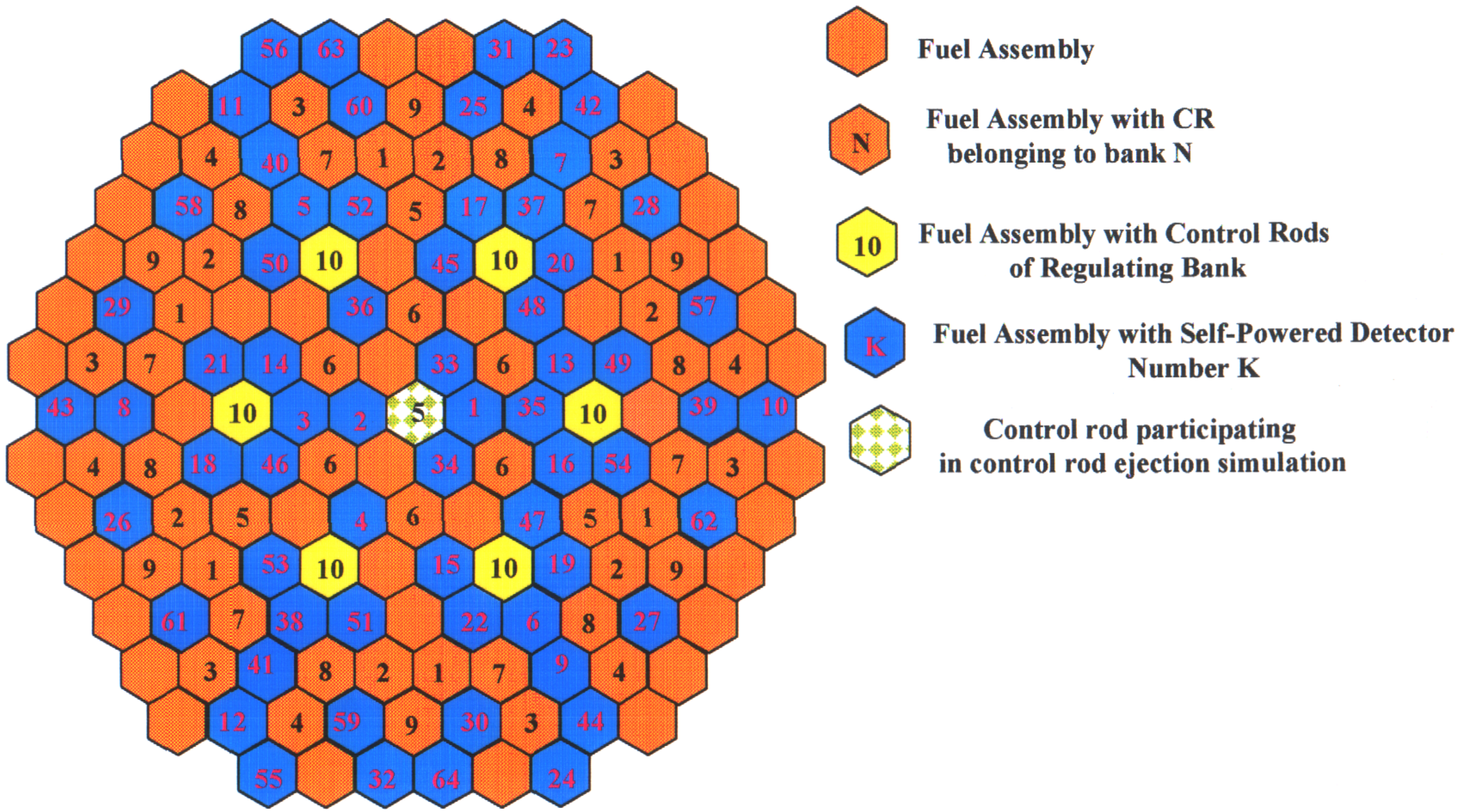
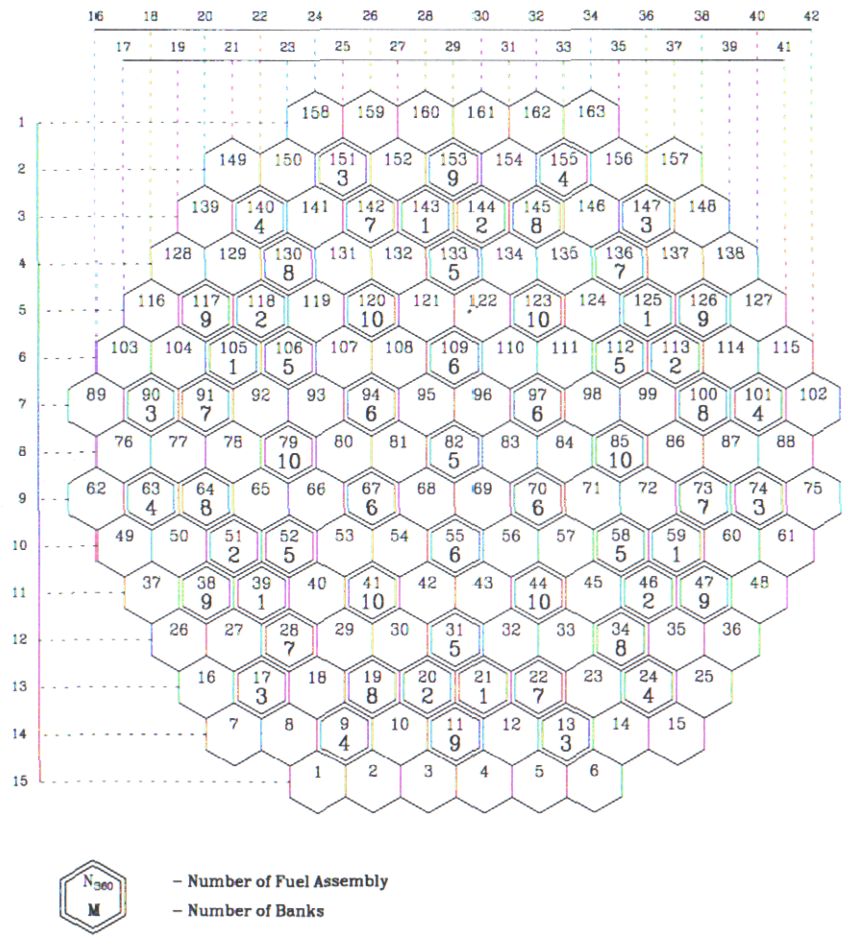


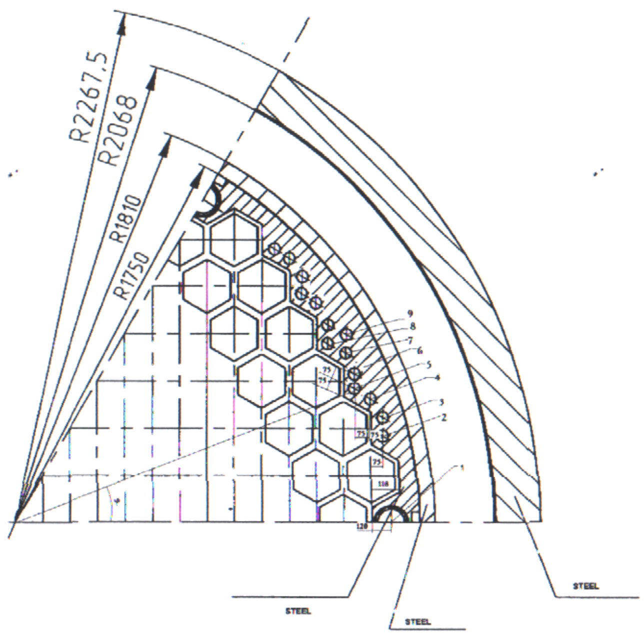
Fig.3-10. Assemblies Numeration in VVER-1000 Core



– Number of Fuel Assembly

– Number of Banks

Fig.3-11. Model of VVER-1000 Reflector



Hole number	Distance from core center (R)	Angle ($\varphi°$)	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. 3-12. Reflector "Assembly" of Type 1

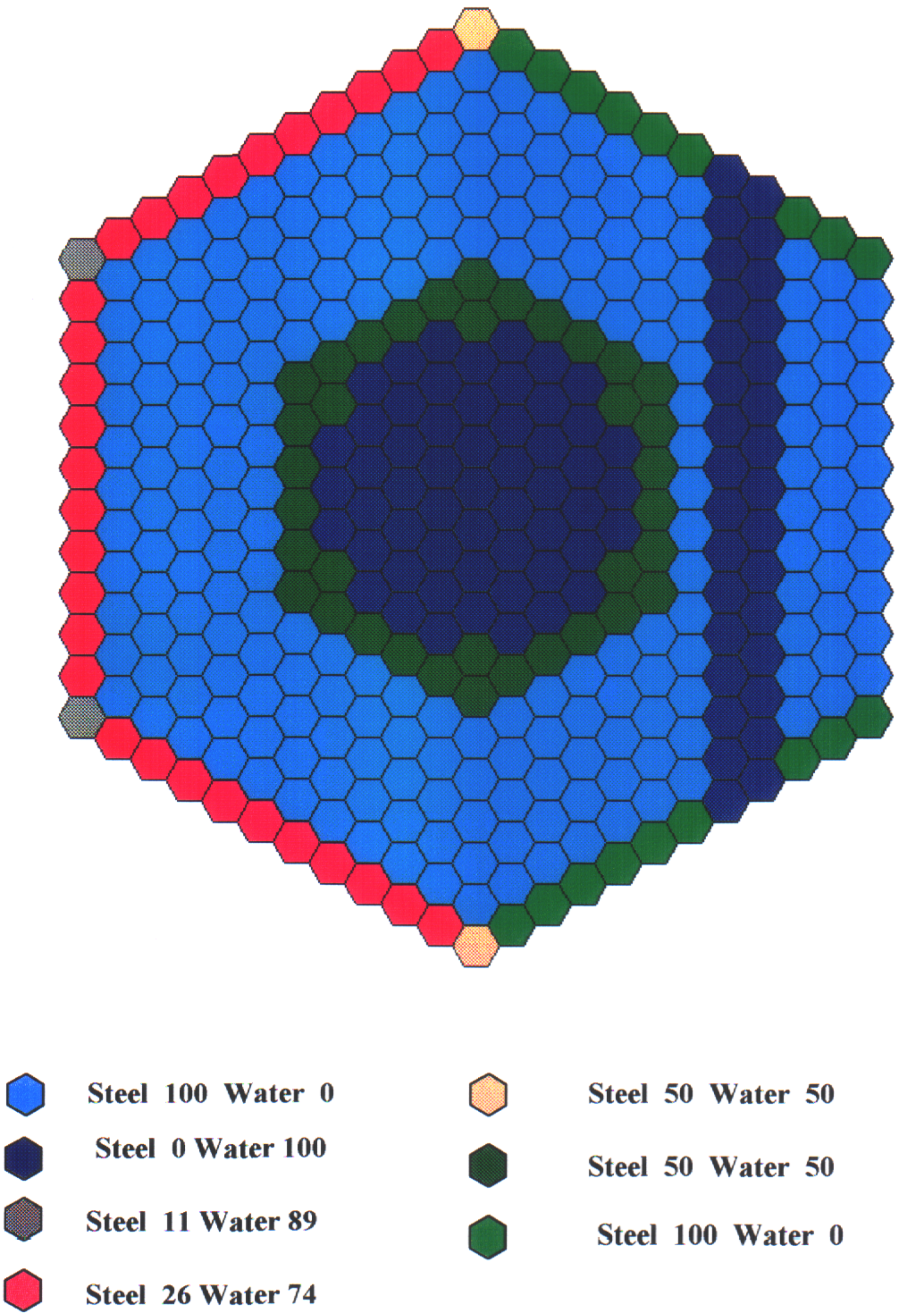


Fig. 3-13. Reflector "Assembly" of Type 2

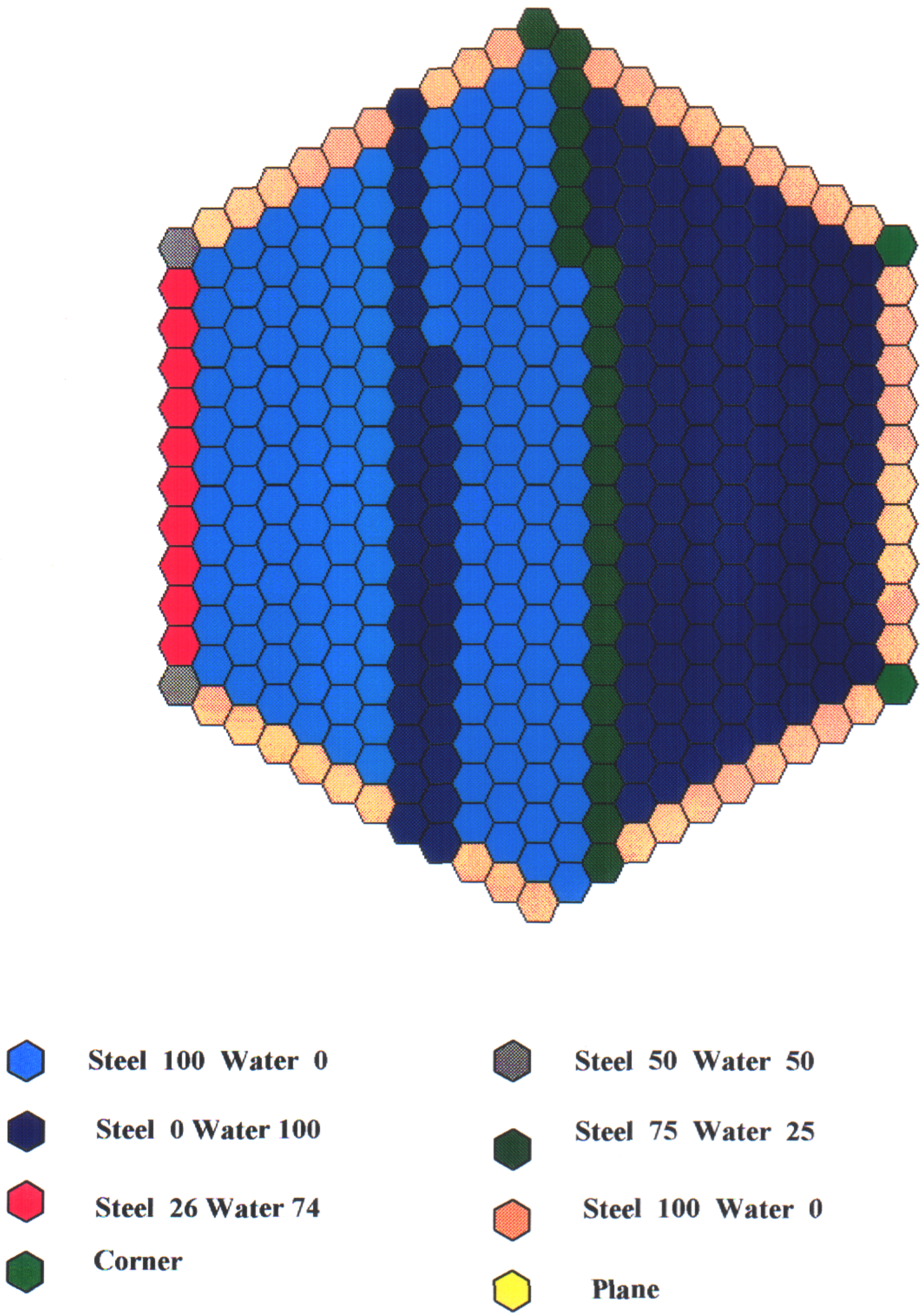


Fig. 3-14. Reflector "Assembly" of Type 3

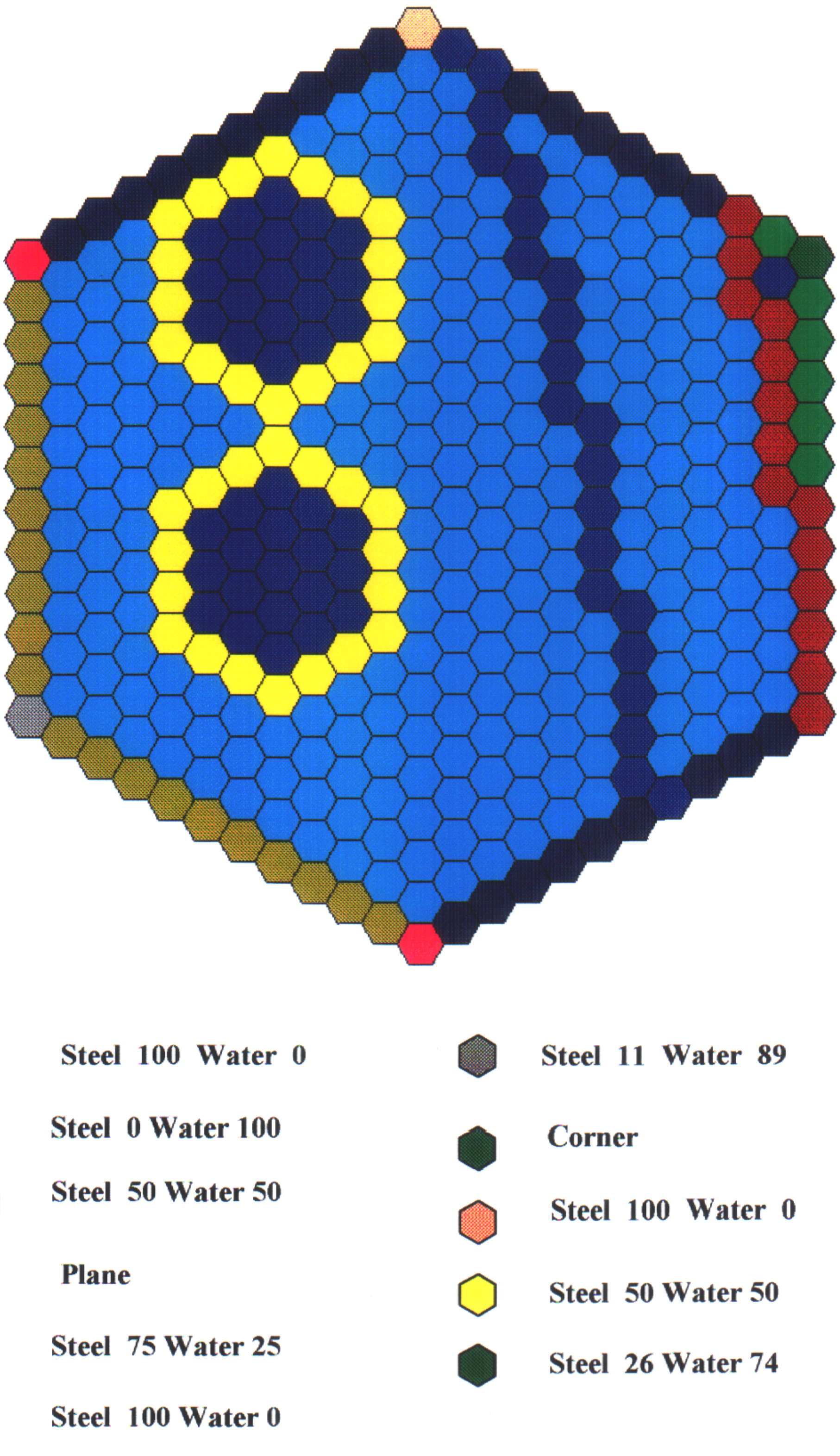


Fig. 3-15. Reflector "Assembly" of Type 4

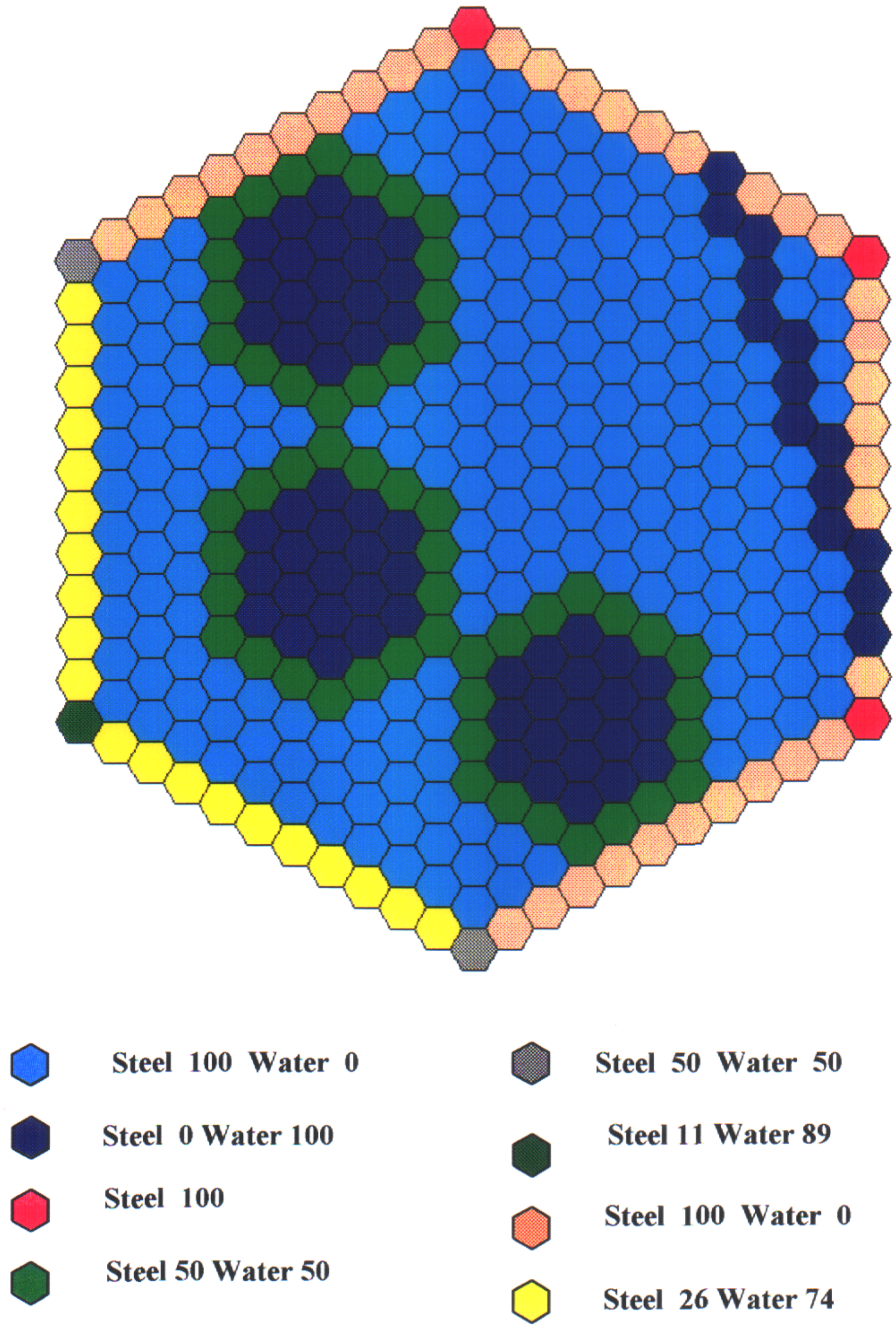


Fig. 3-16. Reflector "Assembly" of Type 5

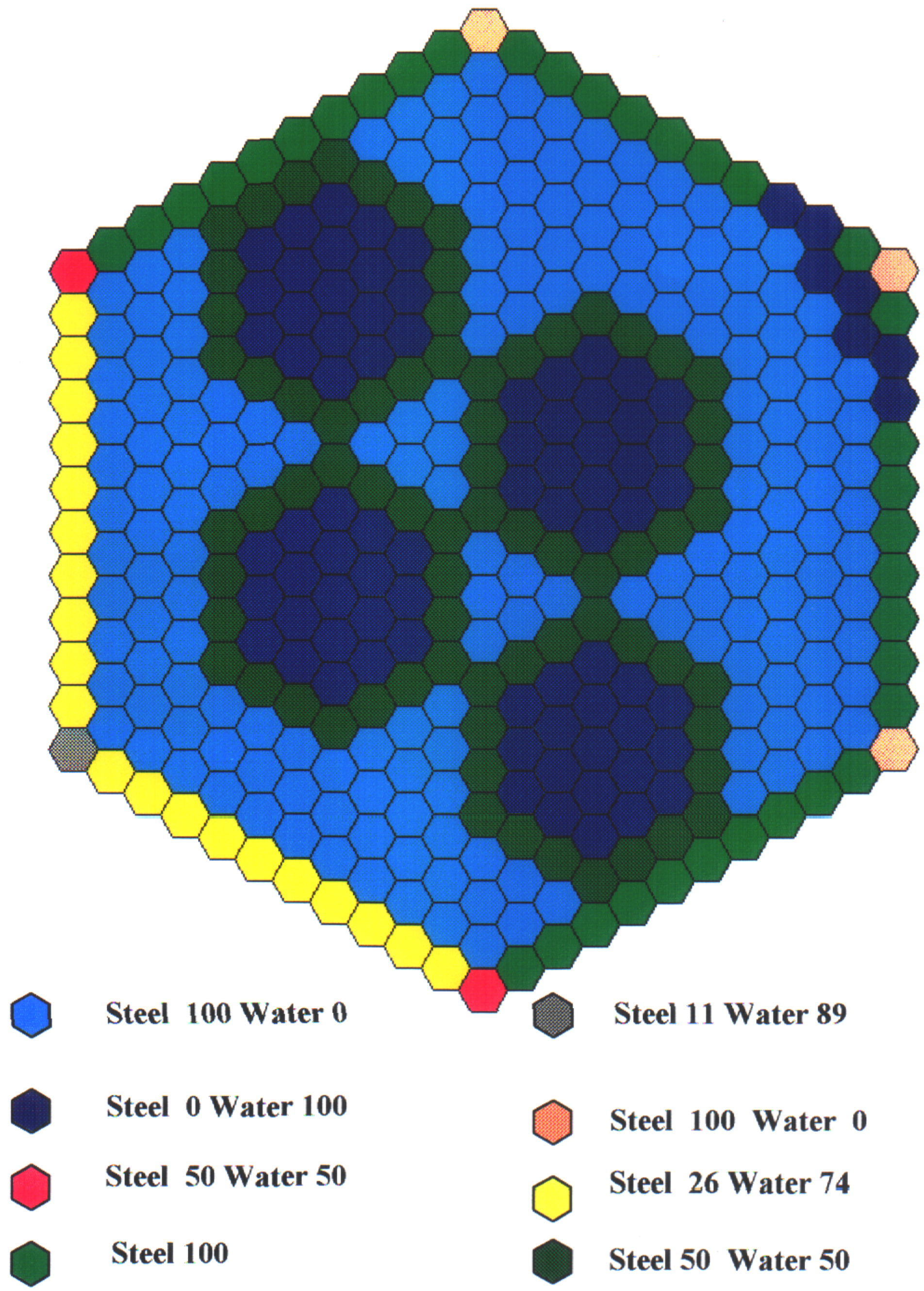
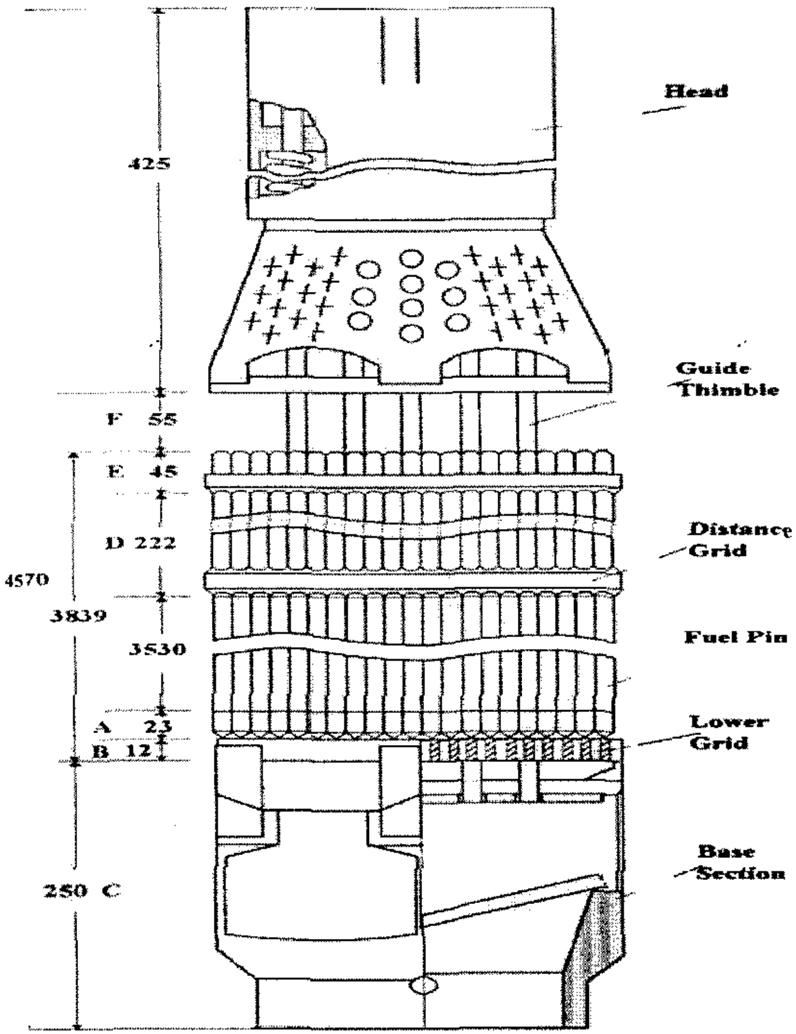


Fig.3-17. Core Design. Material Contents (Vol%) in Axial Direction

Zone	H_2O	St.-Steel	Zr	He
A	58.0	0	42.0	0
B	57.0	33.0	10.0	0
C	67.0	33.0	0	0
D	56.0	0	13.8	30.2
E	56.0	0	32.5	11.5
F	98.9	0	0	0



4. Central Control Rod Ejection from the VVER-1000 Core

4.1. *Process being investigated*: Central control rod is ejected by pressure drop caused by destroying of the moving mechanism cover.

4.2. *Initial state of the core*:

- EOC for equilibrium refueling regime, see Figures 4-1, 4-2.
- Thermal power of reactor - nominal power level W_{nom}.
- Coolant flow rate - 84 000 m^3/h.
- Positions of control rods regulating bank and central control rod are - 20% from the core bottom.
- Xenon and samarium concentrations - equilibrium values corresponding to the reactor nominal power level.

4.3. *Scenario of transient.*

Central control rod is ejected by pressure drop caused by destroying of the moving mechanism cover. Time point of ejection start corresponds to 0.1 sec. Initial control rod position - 20% from the core bottom. Duration of ejection is 0.05 sec. The scram is actuated at time point 1.0 sec. Duration of the rod drop is 4 sec.

4.4. *Main calculation results.*

For the uranium zone the worth of ejected cluster is 87.4 pcm, and for the MOX zone it is 93.6 pcm.

The calculation results of dynamical processes are shown in Figs. 4-3 - 4-8. Figs. 4-5 - 4-8 present the characteristics of «hot» channel with sampling all over the core.

The «hot» channel power was taken as

$q_{hot.chan.ij}=q_{ij} * K_{eng} * K_k^{max}$,

where q_{ij} is the power averaged over the calculation prism ij, with j being the number of layer in the core height; i - the FA number;

K_{eng} - is the engineering margin coefficient accounting for the errors in the calculation of power averaged over FA cross section, q_{ij}, and of the flow rate through the i-th FA;

K_k^{max} - is the maximum radial peaking coefficient in the hottest FA.

In the present work:

$K_{eng} = 1.16$,

$K_k = 1.08$

Concluding it can be said that the transition to MOX fuel does not noticeably alter the dynamical process.

FA NUMBER
BURNUP(KG F-P/T FUEL)
RELATIVE POWER
TEMPERATURE DROP

28
3.7622E+01
4.0063E-01
1.2944E+01

26	27
1.9201E+01	7.1680E+00
1.1399E+00	8.7605E-01
3.6828E+01	2.8304E+01

23	24	25
2.6445E+01	1.0807E+01	3.4939E+01
1.2528E+00	1.2681E+00	6.1422E-01
4.0476E+01	4.0970E+01	1.9845E+01

19	20	21	22
1.2328E+01	3.2896E+01	1.1496E+01	3.5234E+01
1.2386E+00	1.1472E+00	1.2955E+00	6.1234E-01
4.0019E+01	3.7063E+01	4.1857E+01	1.9784E+01

14	15	16	17	18
3.8448E+01	2.5599E+01	3.2967E+01	1.0780E+01	7.2127E+00
8.8331E-01	1.1508E+00	1.1438E+00	1.2664E+00	8.8160E-01
2.8539E+01	3.7181E+01	3.6954E+01	4.0914E+01	2.8483E+01

8	9	10	11	12	13
2.7236E+01	3.8225E+01	1.2296E+01	2.7084E+01	1.9293E+01	3.3842E+01
1.0107E+00	8.8599E-01	1.2362E+00	1.2394E+00	1.1420E+00	4.2781E-01
3.2653E+01	2.8625E+01	3.9939E+01	4.0044E+01	3.6898E+01	1.3822E+01

1	2	3	4	5	6	7
3.9690E+01	2.7321E+01	1.1944E+01	2.5385E+01	2.6401E+01	1.1301E+01	3.4976E+01
5.8580E-01	9.5214E-01	1.0889E+00	8.3411E-01	1.1438E+00	1.2963E+00	6.4059E-01
1.8926E+01	3.0762E+01	3.5181E+01	2.6949E+01	3.6953E+01	4.1880E+01	2.0696E+01

Fig. 4-1. Uranium zone. Assembly burnup, relative power and temperature drop for control rod ejection calculation.

FA NUMBER
BURNUP(KG F-P/T FUEL)
RELATIVE POWER
TEMPERATURE DROP

28
3.4268E+01
3.9600E-01
1.2794E+01

26	27
1.8990E+01	6.9459E+00
1.1223E+00	8.1328E-01
3.6260E+01	2.6276E+01

23	24	25
2.5186E+01	1.0306E+01	3.3242E+01
1.2638E+00	1.2065E+00	5.8453E-01
4.0833E+01	3.8981E+01	1.8885E+01

19	20	21	22
1.2362E+01	3.8044E+01	1.1232E+01	3.5261E+01
1.2459E+00	1.0365E+00	1.2521E+00	5.6817E-01
4.0254E+01	3.3488E+01	4.0454E+01	1.8357E+01

14	15	16	17	18
3.2580E+01	2.5533E+01	3.9117E+01	1.0336E+01	6.9199E+00
1.0020E+00	1.1177E+00	9.9172E-01	1.2165E+00	8.1284E-01
3.2373E+01	3.6110E+01	3.2041E+01	3.9302E+01	2.6262E+01

8	9	10	11	12	13
2.7549E+01	3.2546E+01	1.2436E+01	2.5278E+01	1.8951E+01	3.3448E+01
1.0752E+00	1.0027E+00	1.2582E+00	1.2760E+00	1.1249E+00	4.0190E-01
3.4737E+01	3.2394E+01	4.0649E+01	4.1227E+01	3.6343E+01	1.2985E+01

1	2	3	4	5	6	7
4.3063E+01	2.7234E+01	1.2595E+01	2.7219E+01	2.7758E+01	1.1983E+01	3.4088E+01
5.9729E-01	1.0229E+00	1.1730E+00	8.5417E-01	1.2380E+00	1.3372E+00	6.7321E-01
1.9298E+01	3.3049E+01	3.7898E+01	2.7597E+01	3.9997E+01	4.3203E+01	2.1751E+01

Fig. 4-2. MOX zone. Assembly burnup, relative power and temperature drop for control rod ejection calculation

Russian Research Center "Kurchatov Institute"
Spatial Kinetics Calculations of MOX Fuelled Core. Variant 22

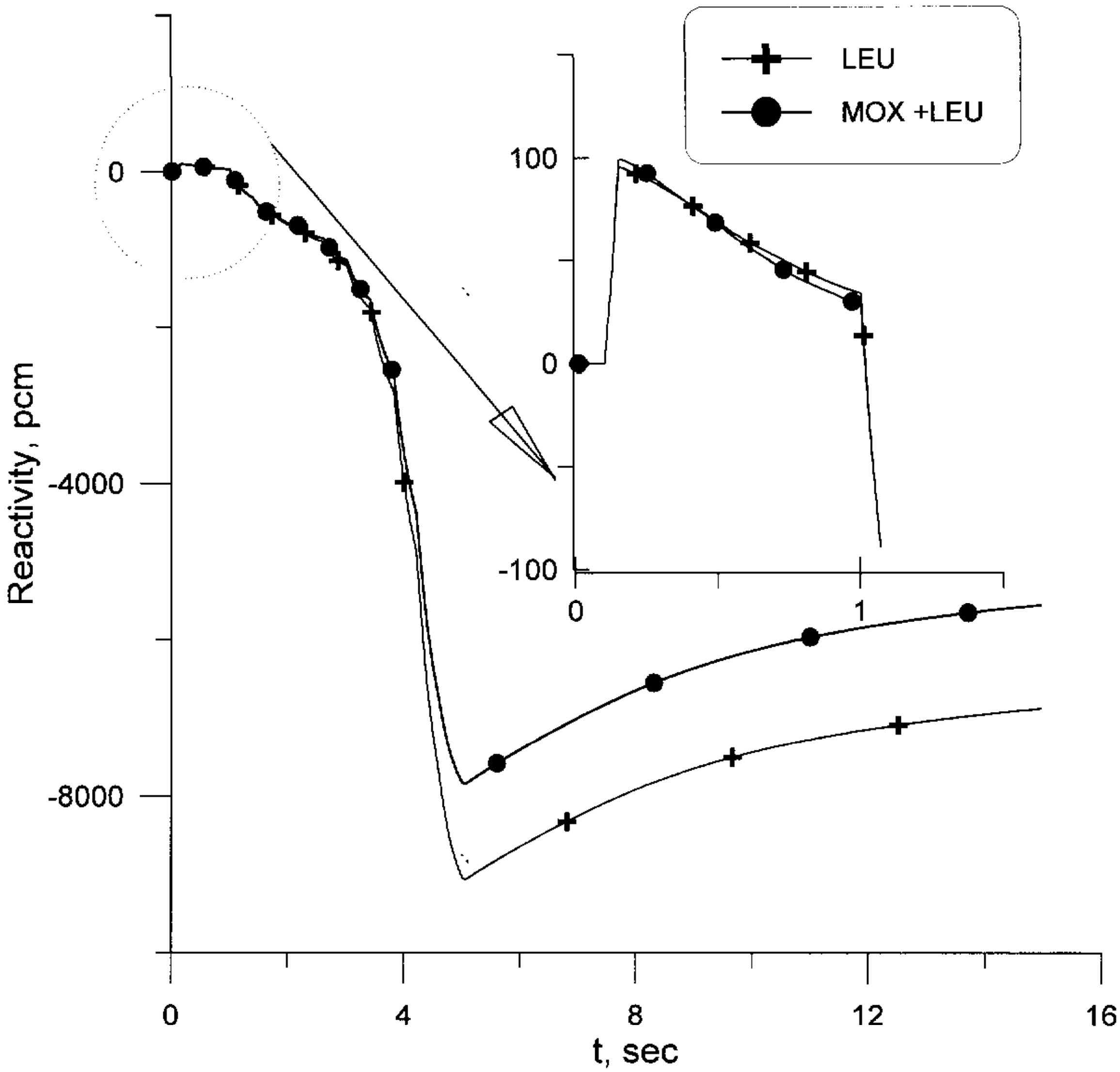


Fig. 4-3. Reactivity. Central Control Rod Ejection

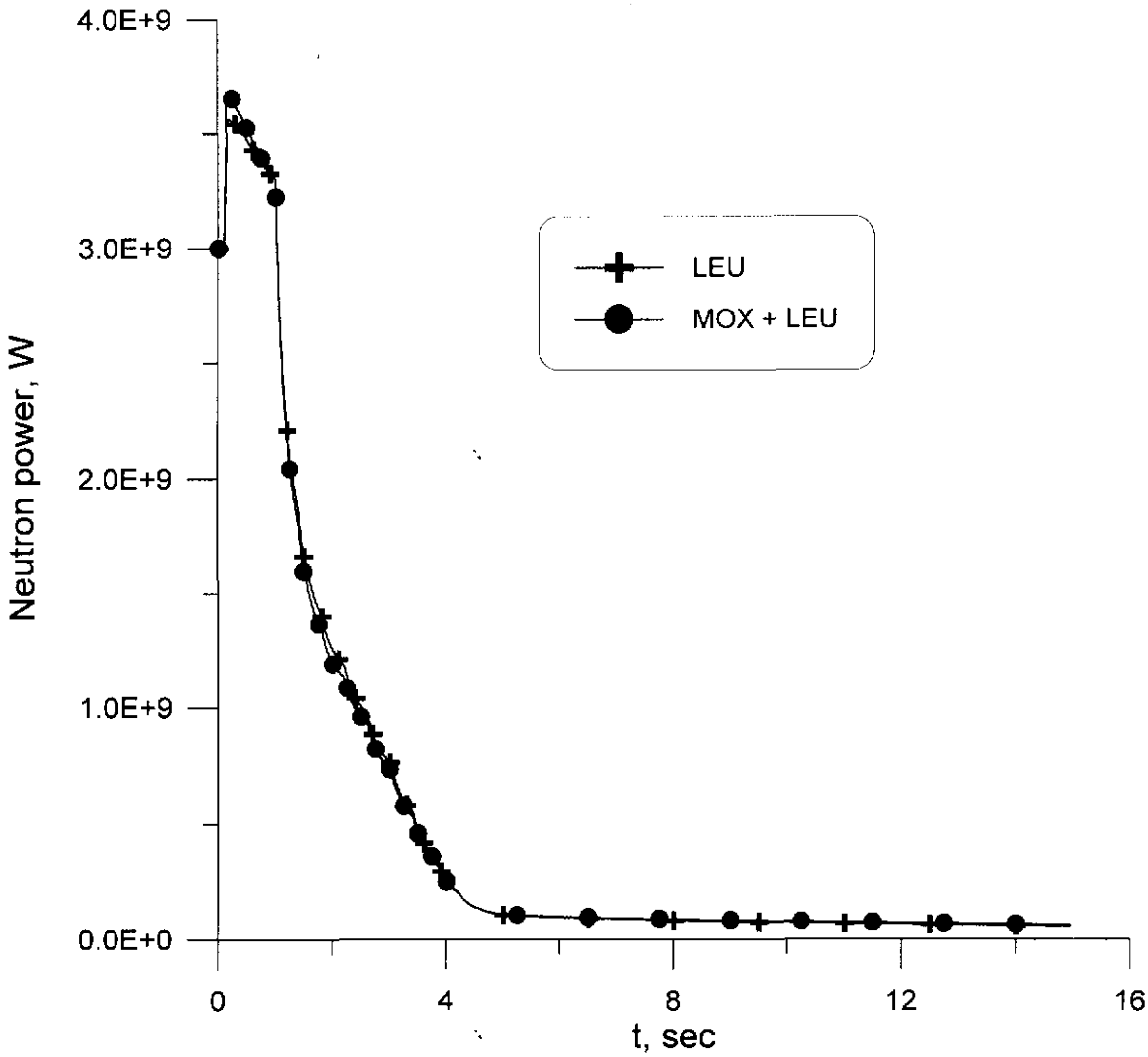


Fig. 4-4. Neutron Power. Central Control Rod Ejection

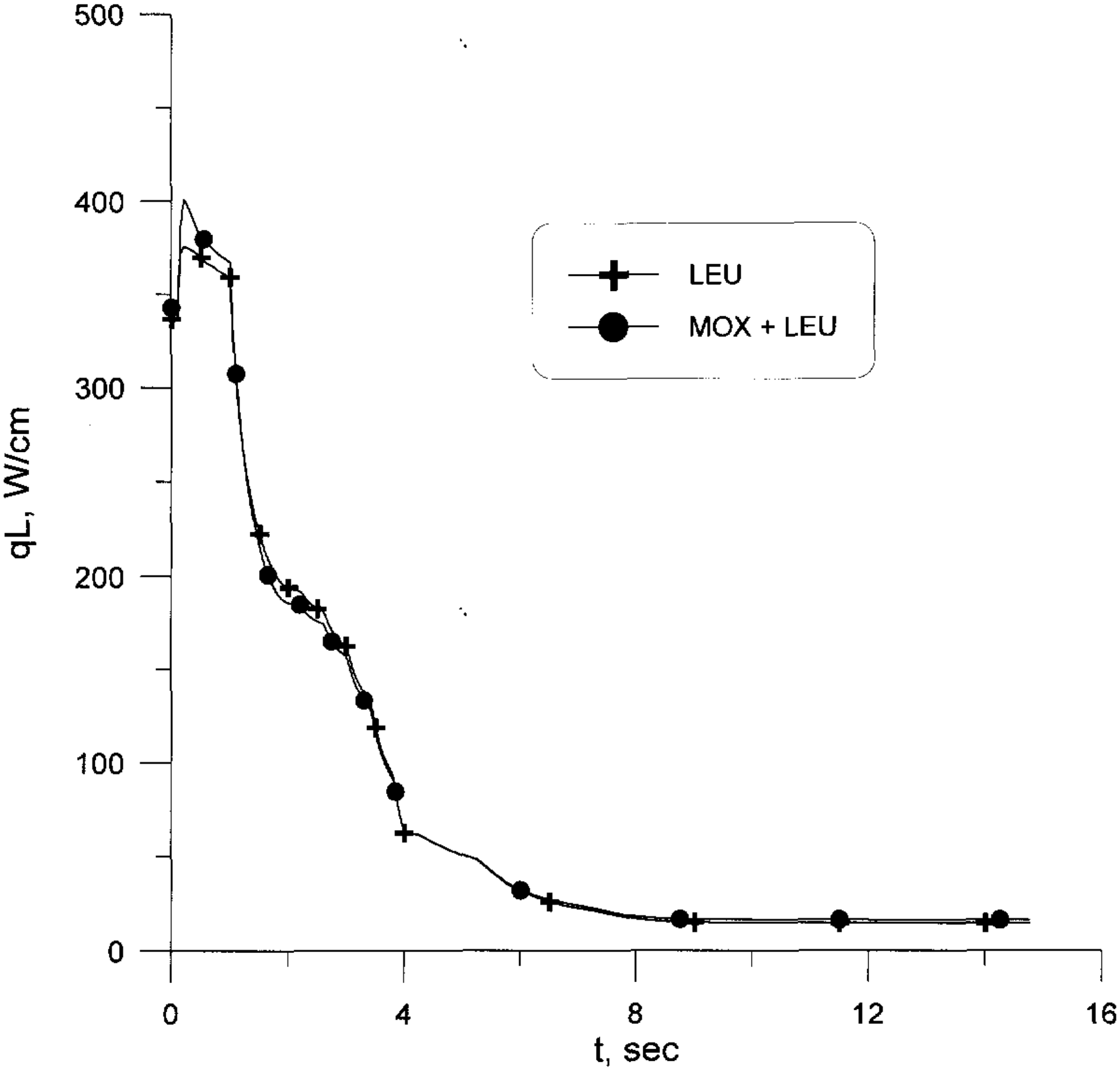


Fig. 4-5. Maximal Neutron Linear Power. Central Control Rod Ejection

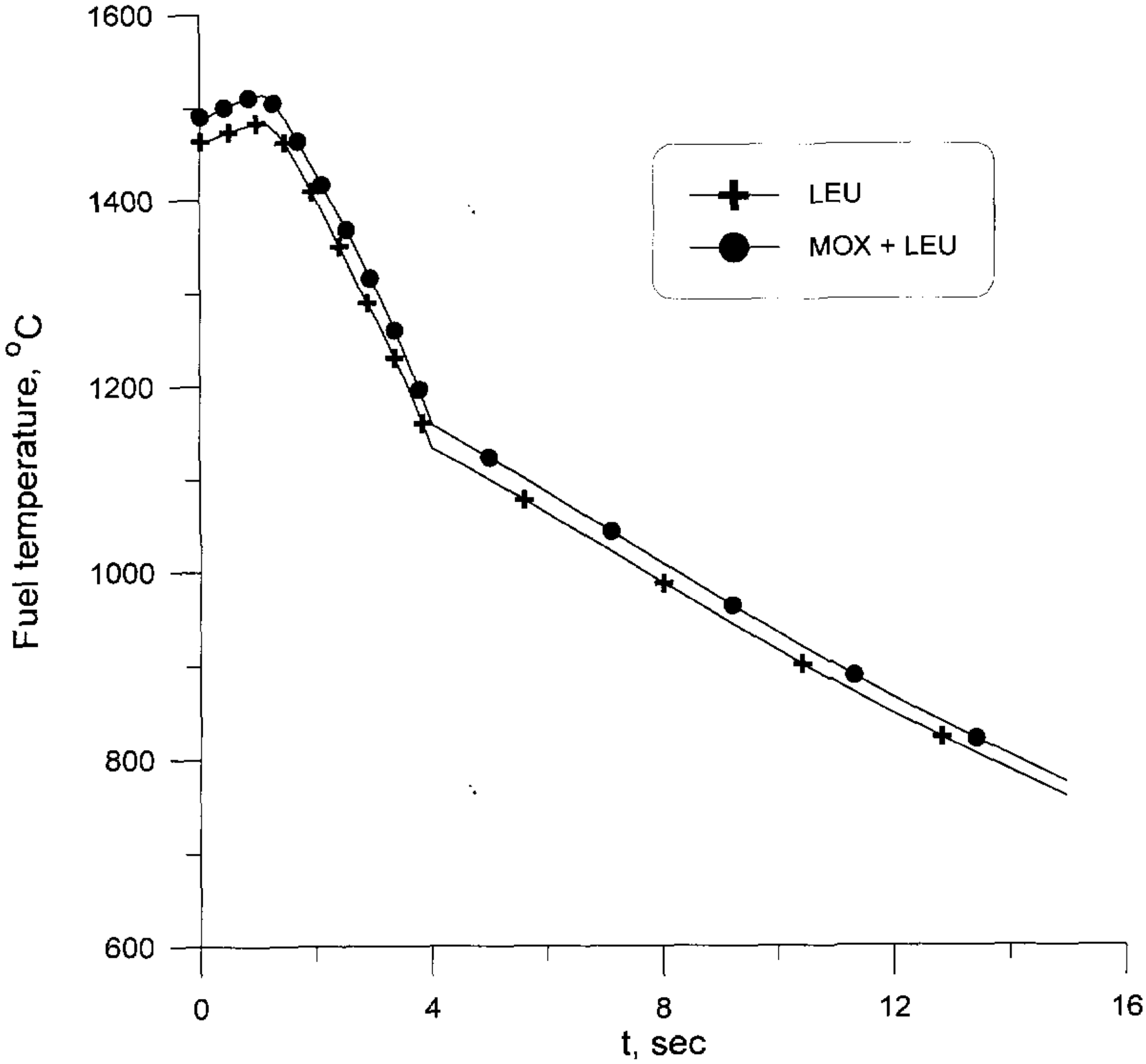


Fig. 4-6. Maximal Fuel Temperature. Central Control Rod Ejection

Russian Research Center "Kurchatov Institute"
Spatial Kinetics Calculations of MOX Fuelled Core. Variant 22

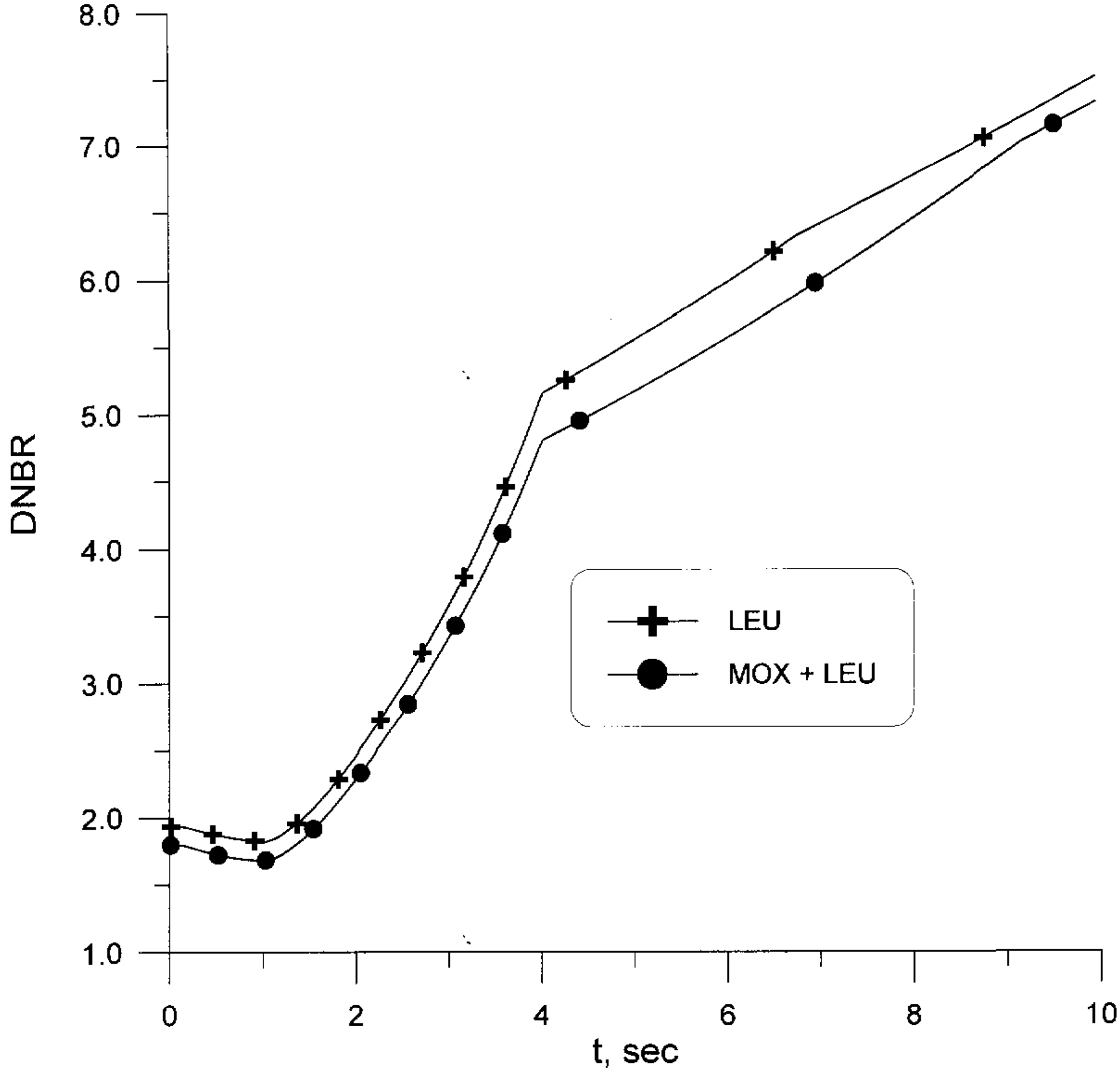


Fig. 4-7. Minimal DNBR. Central Control Rod Ejection

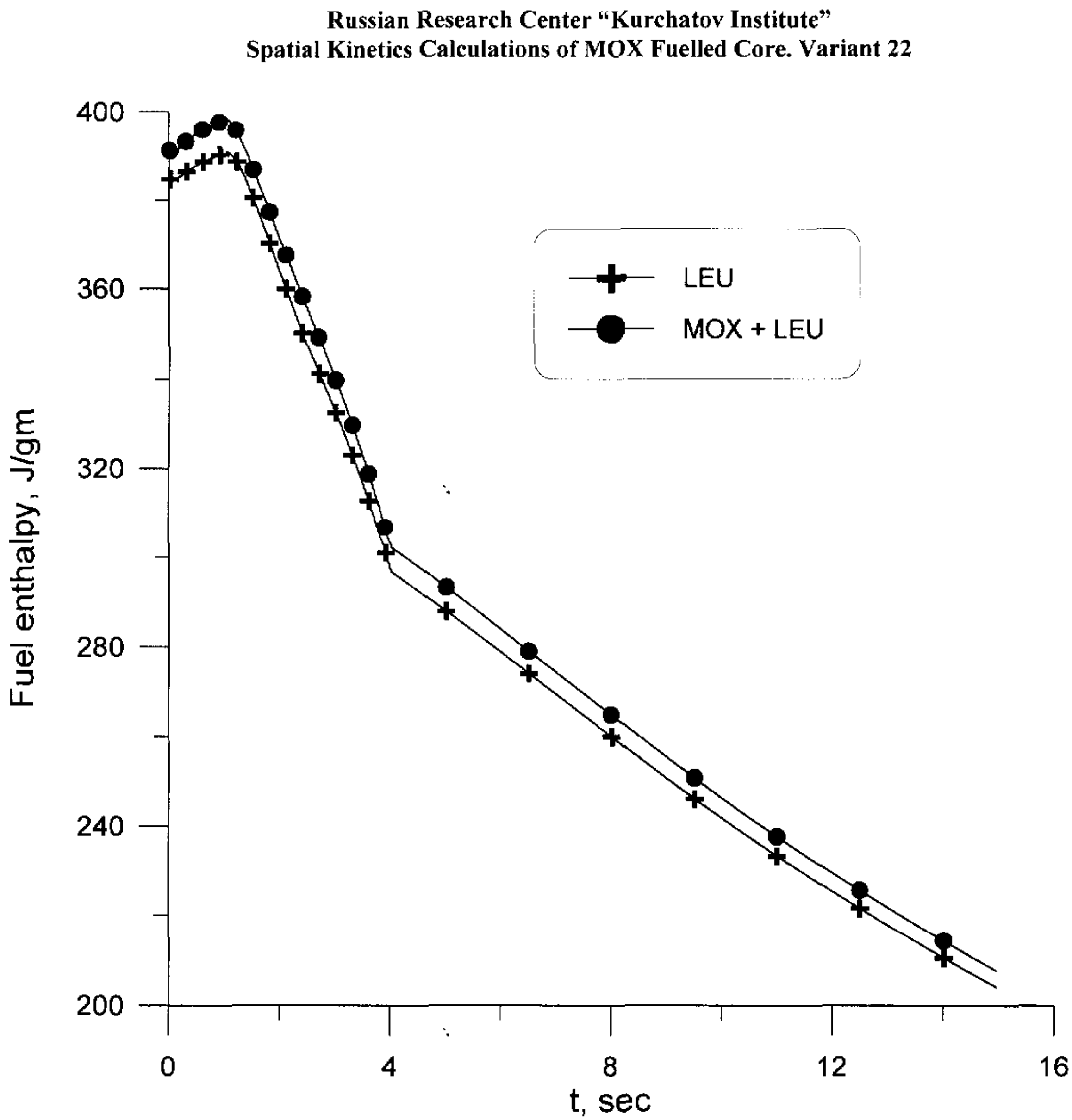


Fig. 4-8. Maximal Fuel Enthalpy. Central Control Rod Ejection

5. Core Cooling in Case of Steam Line Rupture

5.1. *Process being investigated*: Overcooling of the reactor core caused by steam line rupture and non-closure of steam generator stop valve.

5.2. *Initial state of the core*:

- EOC for equilibrium refueling regime.
- W=Wnom.
- Inlet pressure of coolant - 16.4 MPa.
- Coolant flow rate - 84 000 m^3/h.
- Position of control rods regulating bank - 70% from the core bottom.
- Xe eq (Wnom) and Sm eq

5.3. *Scenario of transient*

Accident with steam line rupture in one of the loops between the steam generator and steam generator stop valve, with the power unit de-energized. Accident sequence is shown in Table 5.1.

Table 5-1. Accident sequence

Time, sec	Events and parameters
0.0	The reactor plant operates at Wnom; parameters of primary and secondary circuits are within the permissible limits with allowance for deviations.
0.3	The steam line is depressurized, in the period from 0.3s to 0.6s the rupture cross section increases linearly up to the full steam line cross section.
2.6	Scram is actuated (all rods except for the most effective). Duration of the rod drop is 4 sec.
20	BRUAs operate, the pressure in three intact SG is maintained within the rated limits.

These events lead to variations of core parameters that are shown in Table 5.2. The temperature values are calculated under assumption of the absence of coolant mixing between the loops. Assembly numbers in the one fourth part of the core connected with ruptured steam line are: 4-6, 11-15, 21-25, 31-36, 43-48, 55-61, 69-75, 82. Fig 5.1 shows the changes in the inlet temperature in the overcooled fourth part of the core and other assemblies.

Table 5-2. Variation of Core Parameters in the Accident with Steam Line Rupture and Power Unit de-Energized

Time. sec	Reactor coolant system parameters			
	t_{entry}, °C		Flow rate through the core. G/G_{nom}	Pressure, MPa
	$t_{1/4\ core}$	$t_{3/4\ core}$		
0	287	287	1	16.4
10	272	280	0.8	14.5
20	248	270	0.6	13.9
30	229	259	0.45	13.6
40	220	248	0.35	13.4
50	225	246	0.3	13.4

5.4. *Results of calculations.*

The calculation results of dynamical processes are shown in Figs. 5.1-5.7, Figs. 5.3-5.6 present the characteristics of «hot» channel with sampling all over the core. Fig. 5.7 presents the power of assembly number 24 located in the overcooled fourth part of the core, and power of assembly number 129.

The analysis of Figs. 5.1 – 5.7 shows that for the accident considered the MOX-zone proved to be more dangerous. However this difference is very small.

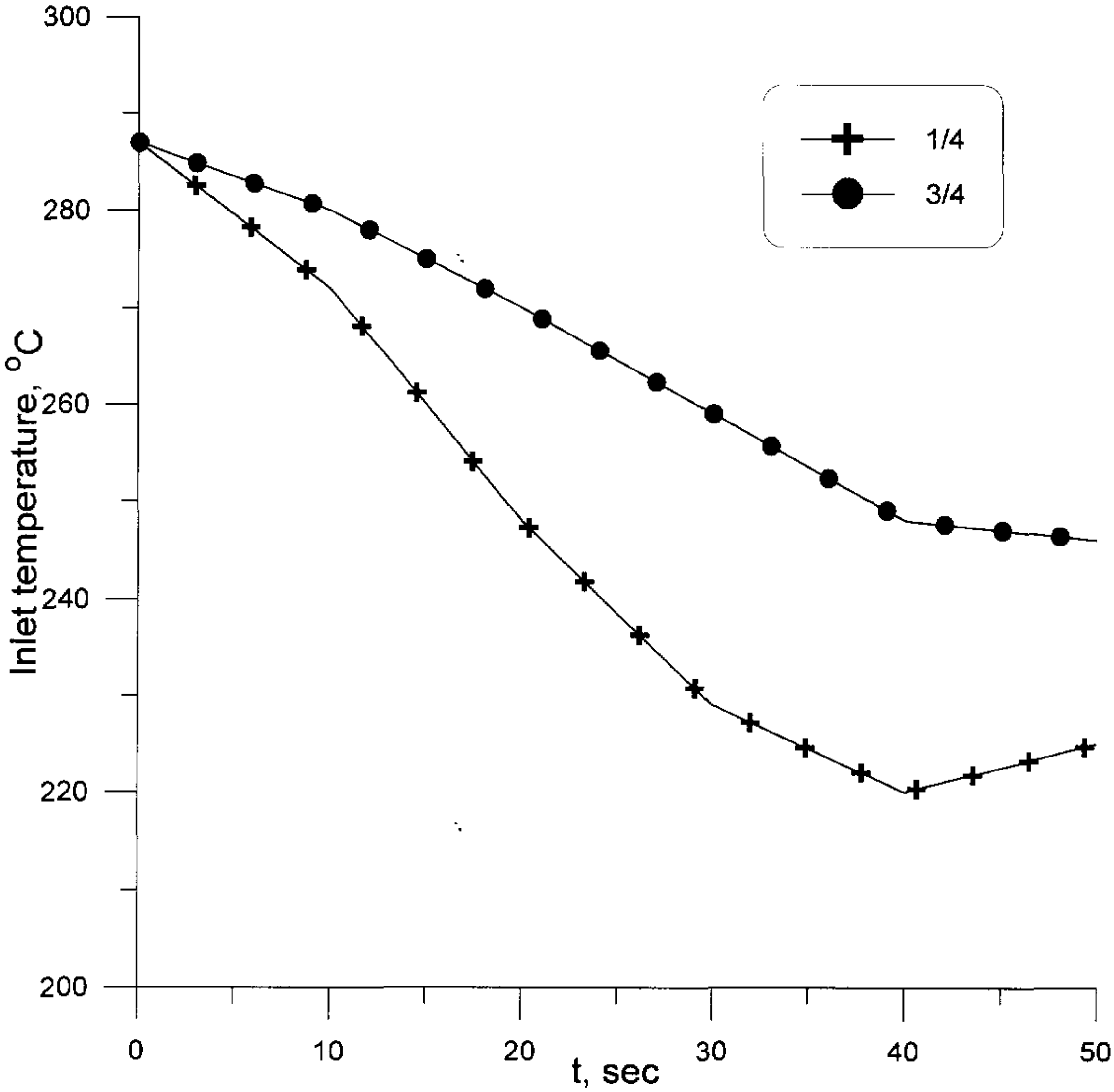


Fig. 5-1. Inlet Temperature in Different Parts of the Core. Partial Core Overcooling

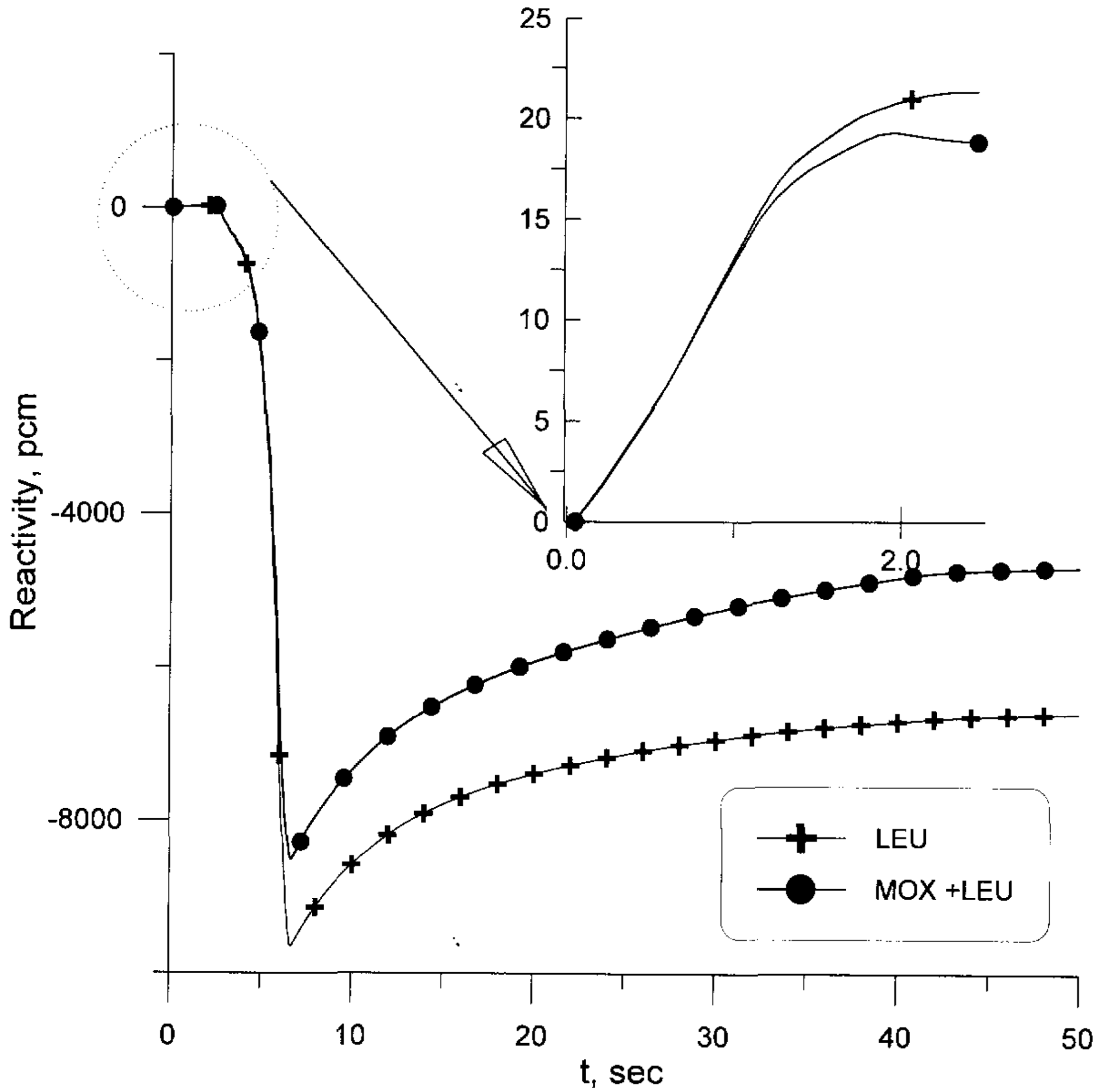


Fig. 5-2. Reactivity. Partial Core Overcooling

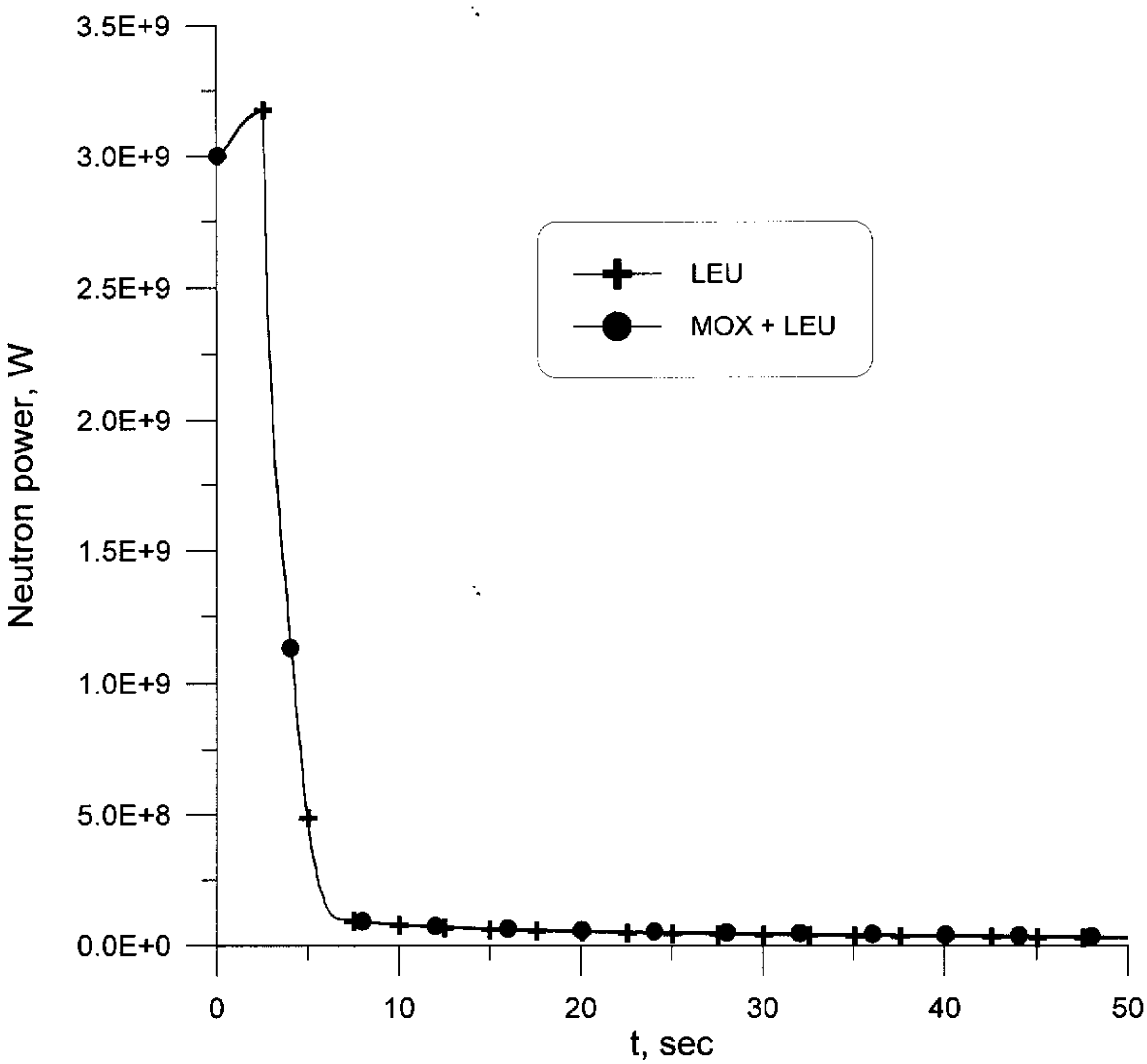


Fig. 5-3. Neutron Power. Partial Core Overcooling

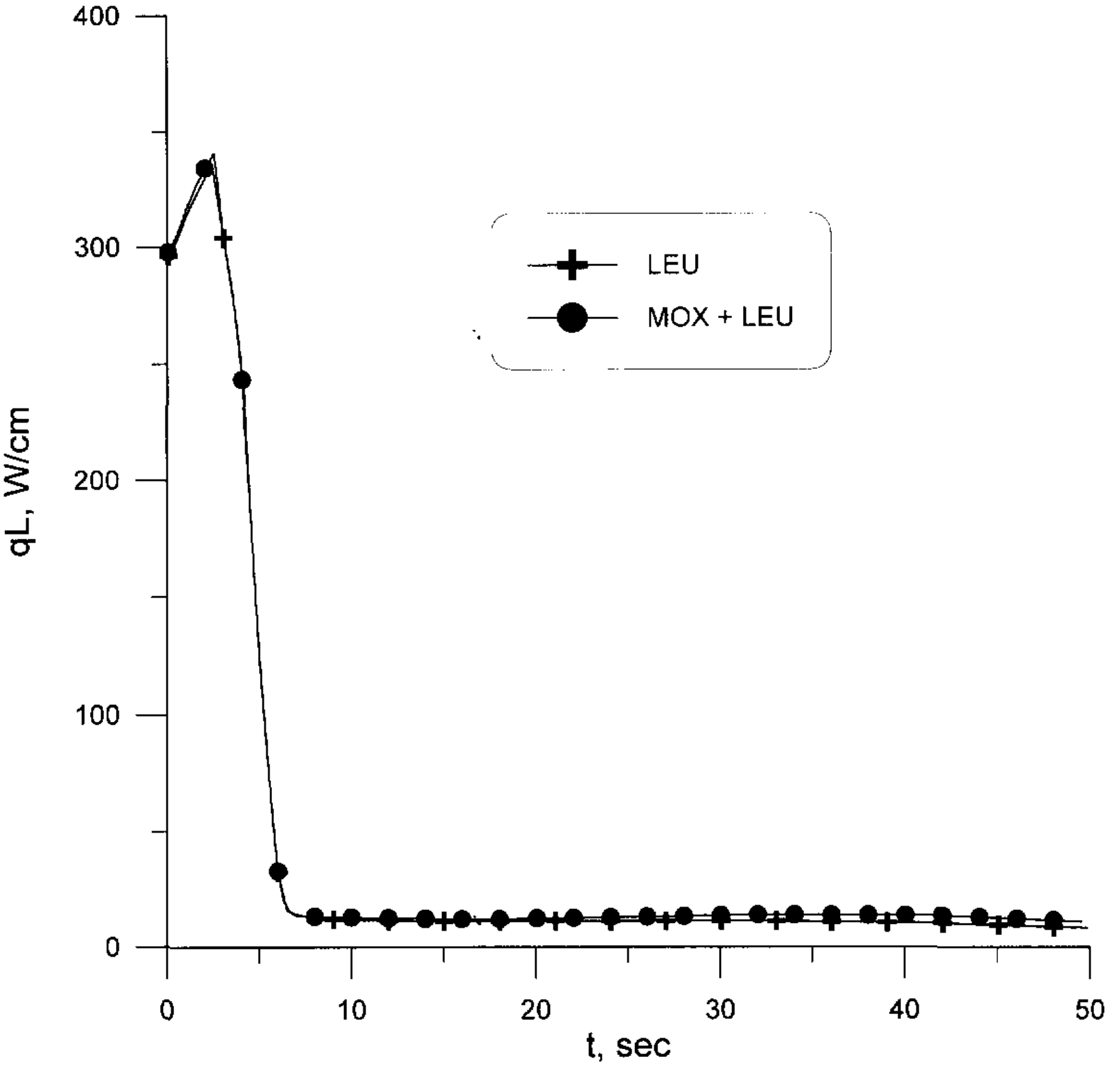


Fig. 5-4. Maximal Neutron Linear Power. Partial Core Overcooling

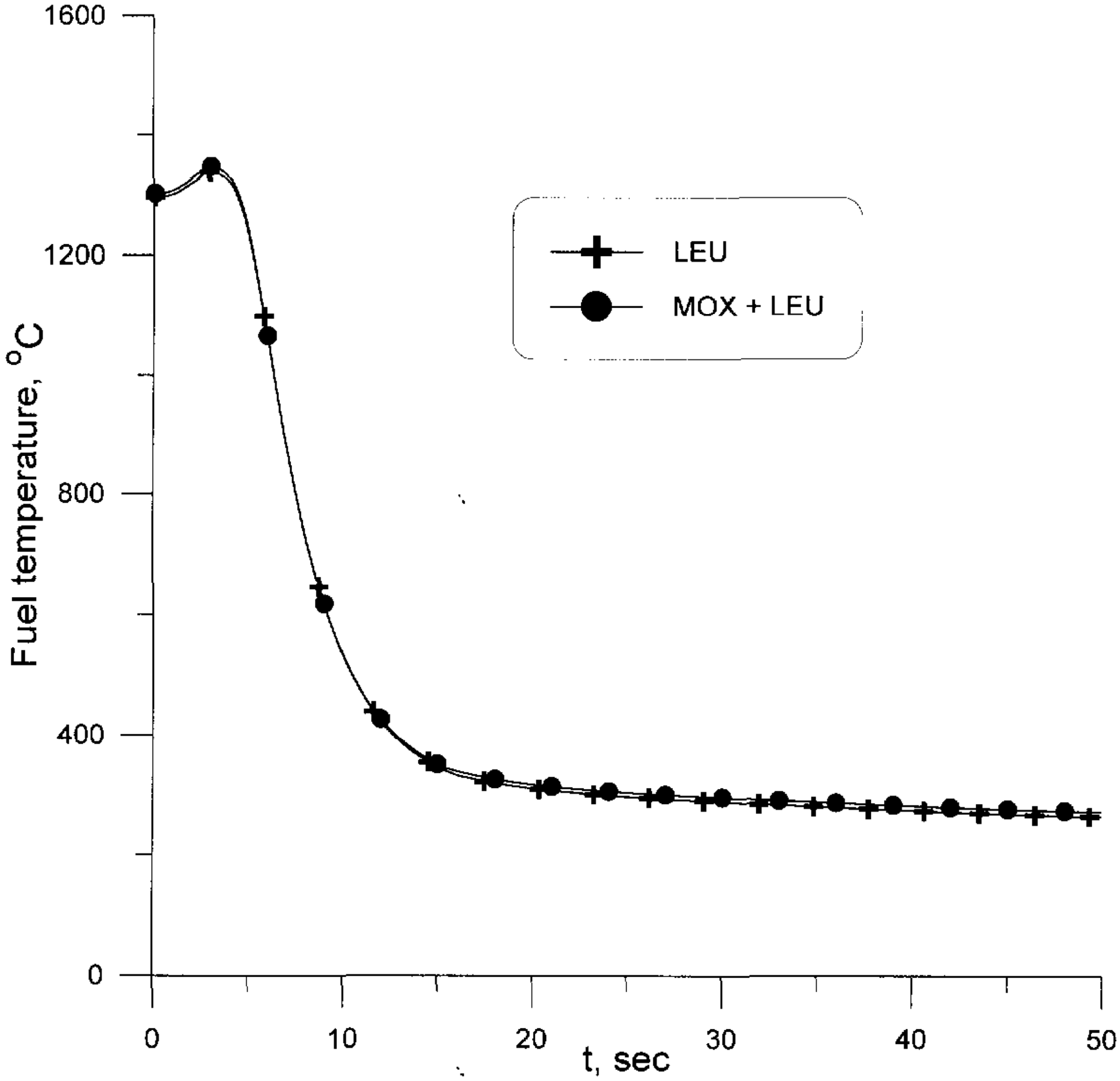


Fig. 5-5. Maximal Fuel Temperature. Partial Core Overcooling

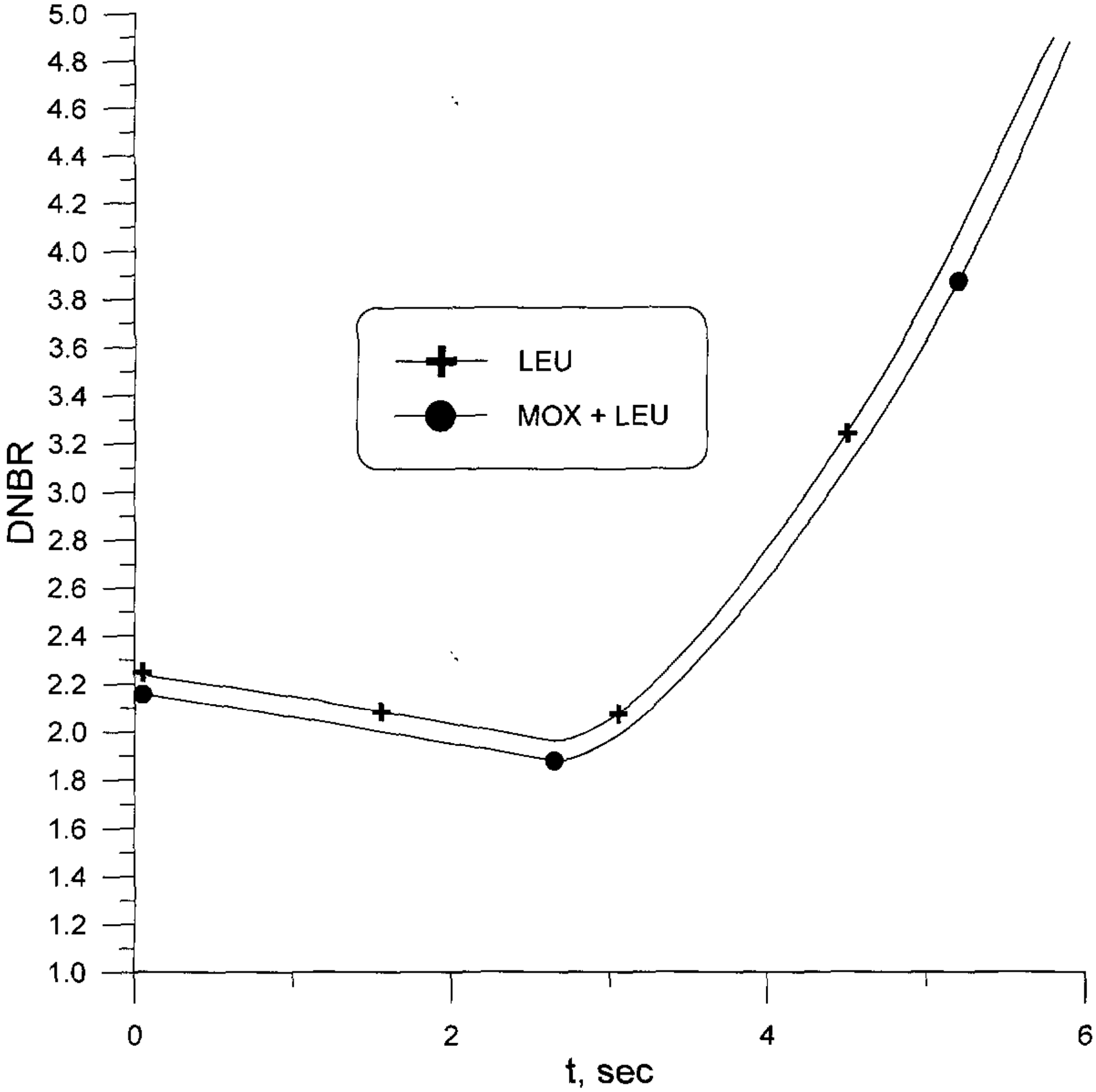


Fig. 5-6. Minimal DNBR. Partial Core Overcooling

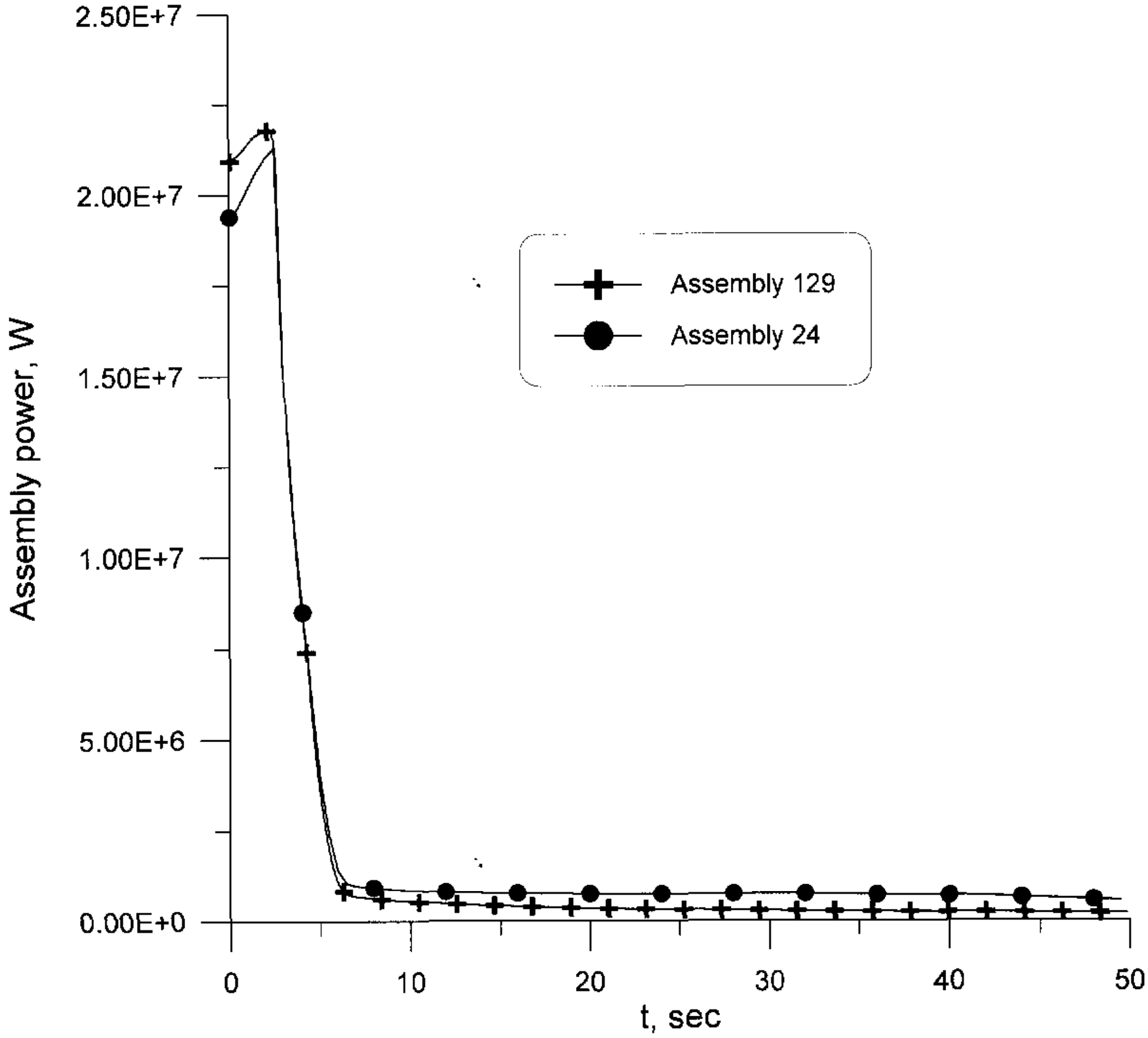


Fig. 5-7. Assembly Power. Partial Core Overcooling. MOX + LEU

6. The Boron Dilution of Coolant in a Part of the VVER-1000 Core

6.1 *Process being investigated*: The boron dilution of coolant in part of the VVER-1000 core caused by penetration of the distillate slug into the core at start up of non-working loop.

6.2. *Initial state of the core*:

- BOC for equilibrium refueling regime.
- W = 0.66 Wnom.
- Three main circulating pumps are working, coolant flow rate - 75% of nominal value 84 000 m^3/h.
- Position of control rods regulating bank - 80% from the core bottom.
- Xe eq (Wnom) and Sm eq.

6.3. *Scenario of transient.*

At initial time point (4.0 sec) the fourth main circulating pump is assumed to be switched on. The flow rate of this pump changes linearly during 30 sec from 0% to 100% of nominal value (21000 m^3/h).

At time point of 6.0 sec the distillate slug comes to the core inlet.

When the reactor power exceeds 110% of initial value, the scram activates. Duration of the rod drop is 4 sec.

Simultaneously with activation of the scram, the most effective control rod located in the cell 58 of the 5^{th} bank is stuck at its upper position.

It is assumed that full amount of coolant from the activated loop comes into the one-fourth part of the core (see below Fig. 6-1).

Within this quarter of the core the distillate and coolant coming from other loops are fully mixed.

The boric acid concentration at inlet of this quarter C(t) changes according to the following dependence:

$$
\begin{cases}
C(t) = C_H; & t < \tau_1 \\
C(t) = \dfrac{3\left(1 - \dfrac{t}{\tau}\right)}{3 + \dfrac{t}{\tau}} C_H; & \tau_1 < t \le \tau \\
C(t) = 0.0; & t > \tau
\end{cases}
$$

where C_H - critical boric acid concentration for initial state;

τ_1 - time point when the distillate slug enters into the core (4 sec);

τ - end of startup period for the 4^{th} pump (34 sec).

Concentration of boric acid in other 3/4 part of the core does not change during the transient.

6.4. *Main calculation results.*

Fig. 6-2 shows the change in the boric acid concentration in the typical fuel assembly of sector 1.

The calculation results of dynamical process are given in Figs. 6-3 - 6-9, Figs. 6-5 - 6-8 show the "hot" channel (sampling is made over the whole core).

As far as the neutron power is concerned, the first downward kink of the curve is accounted for by the drop of safety control rods, the second - for the Doppler effect.

The analysis of Figs. 6-3 – 6-9 shows that for the accident considered the MOX-zone proved to be more dangerous. However this difference is not very important. The short-time DNB both in LEU and MOX + LEU cores is possible.

Fig. 6-1. Numbers of Fuel Assemblies in the Quarter of the Core

82 83 84 85 86 87 88

69 70 71 72 73 74 75

55 56 57 58 59 60 61

43 44 45 46 47 48

32 33 34 35 36
21 22 23 24 25

12 13 14 15

4 5 6 Sector 1

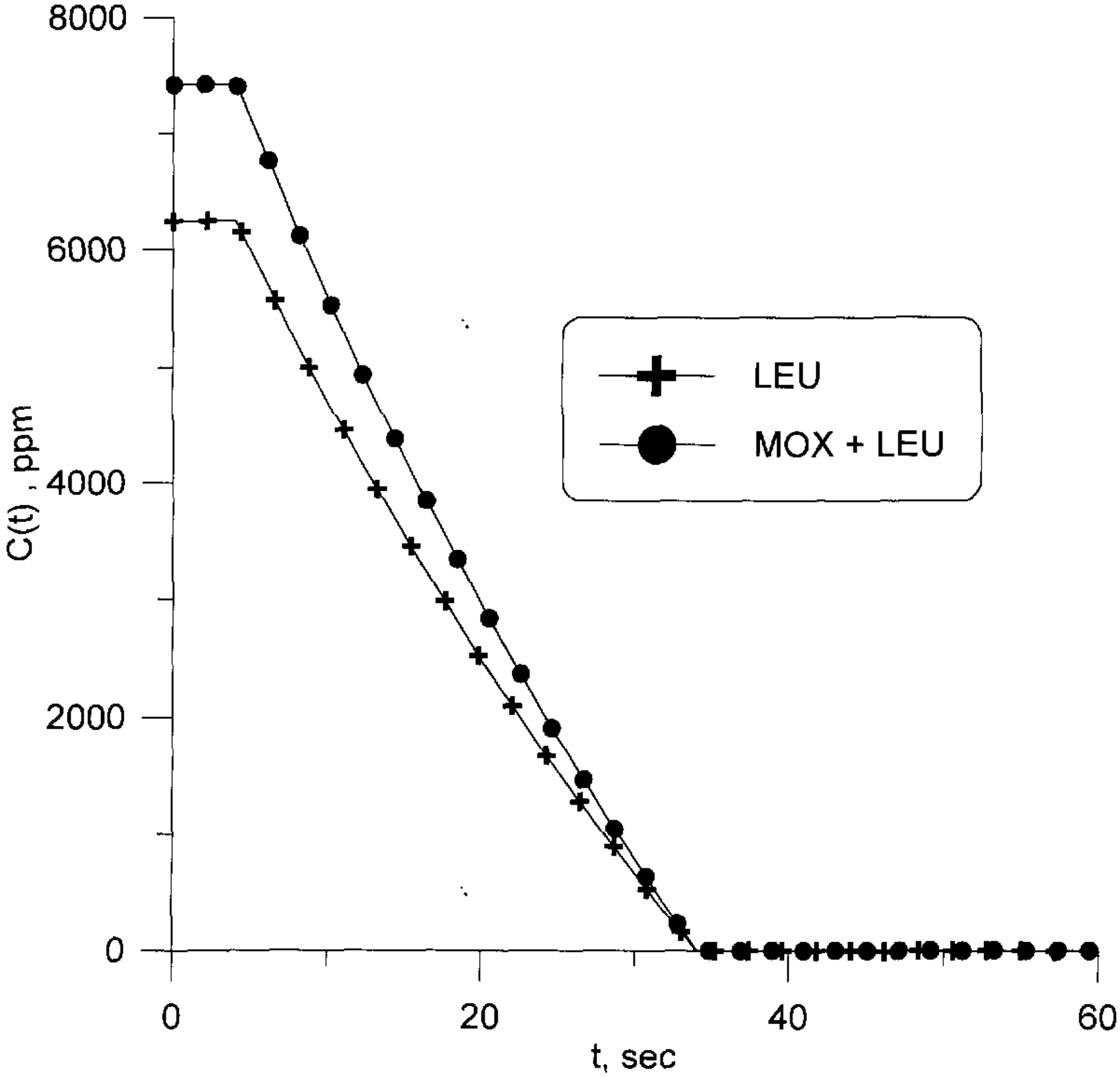


Fig. 6-2. Concentration of Boron Acid. Loop Put into Operation & Boron Dilution

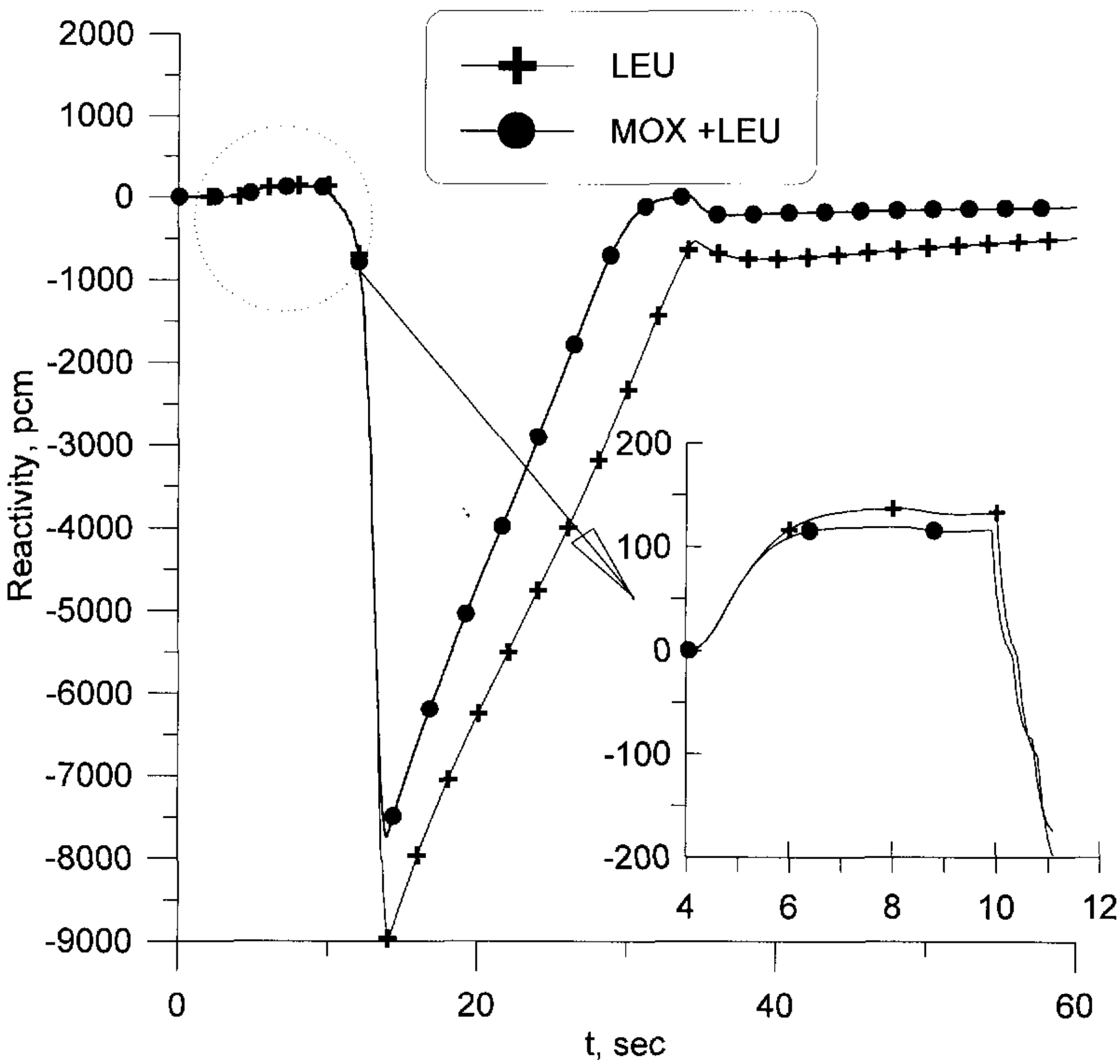


Fig. 6-3. Reactivity. Loop Put into Operation & Boron Dilution

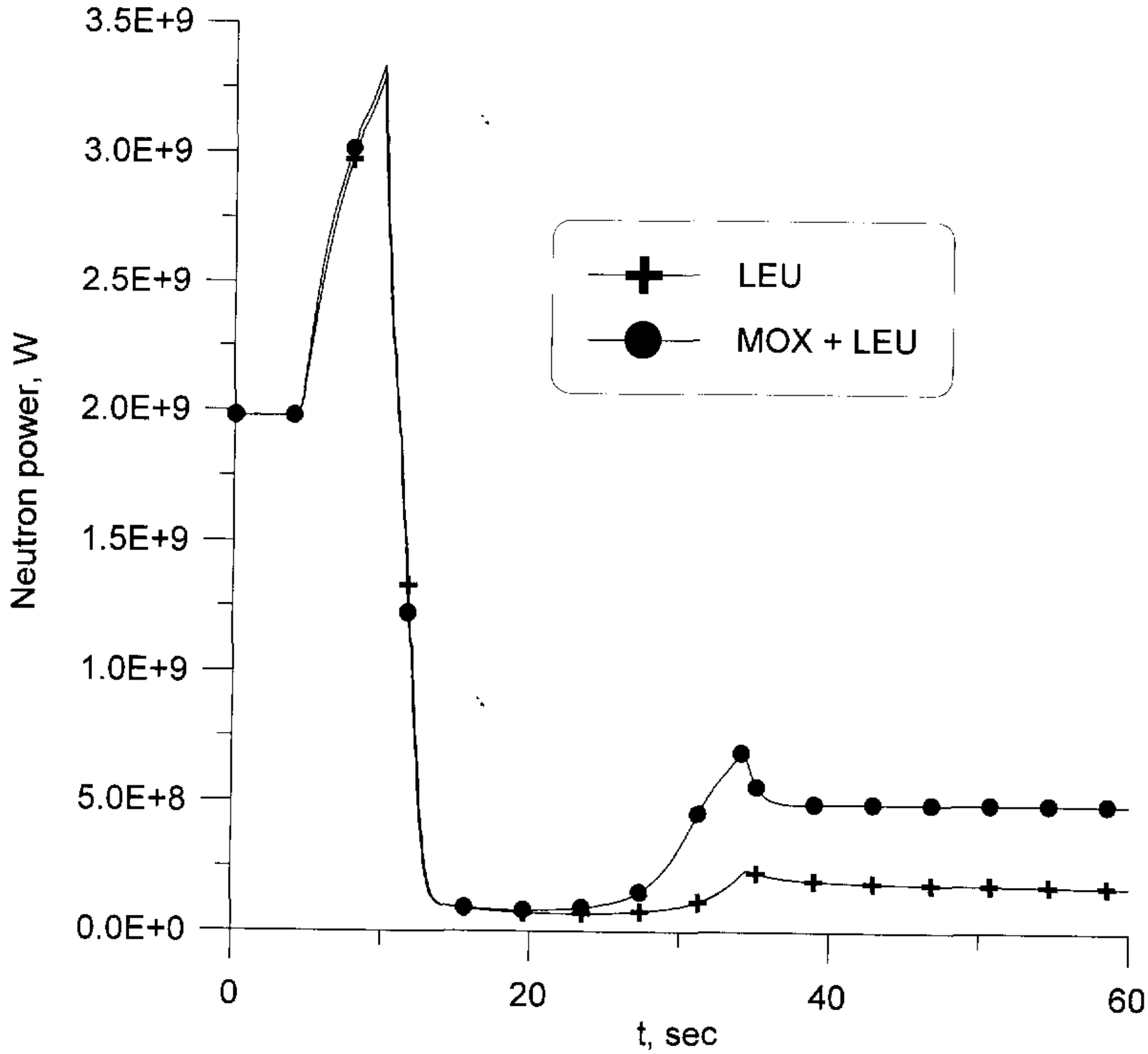


Fig. 6-4. Neutron Power. Loop Put into Operation & Boron Dilution

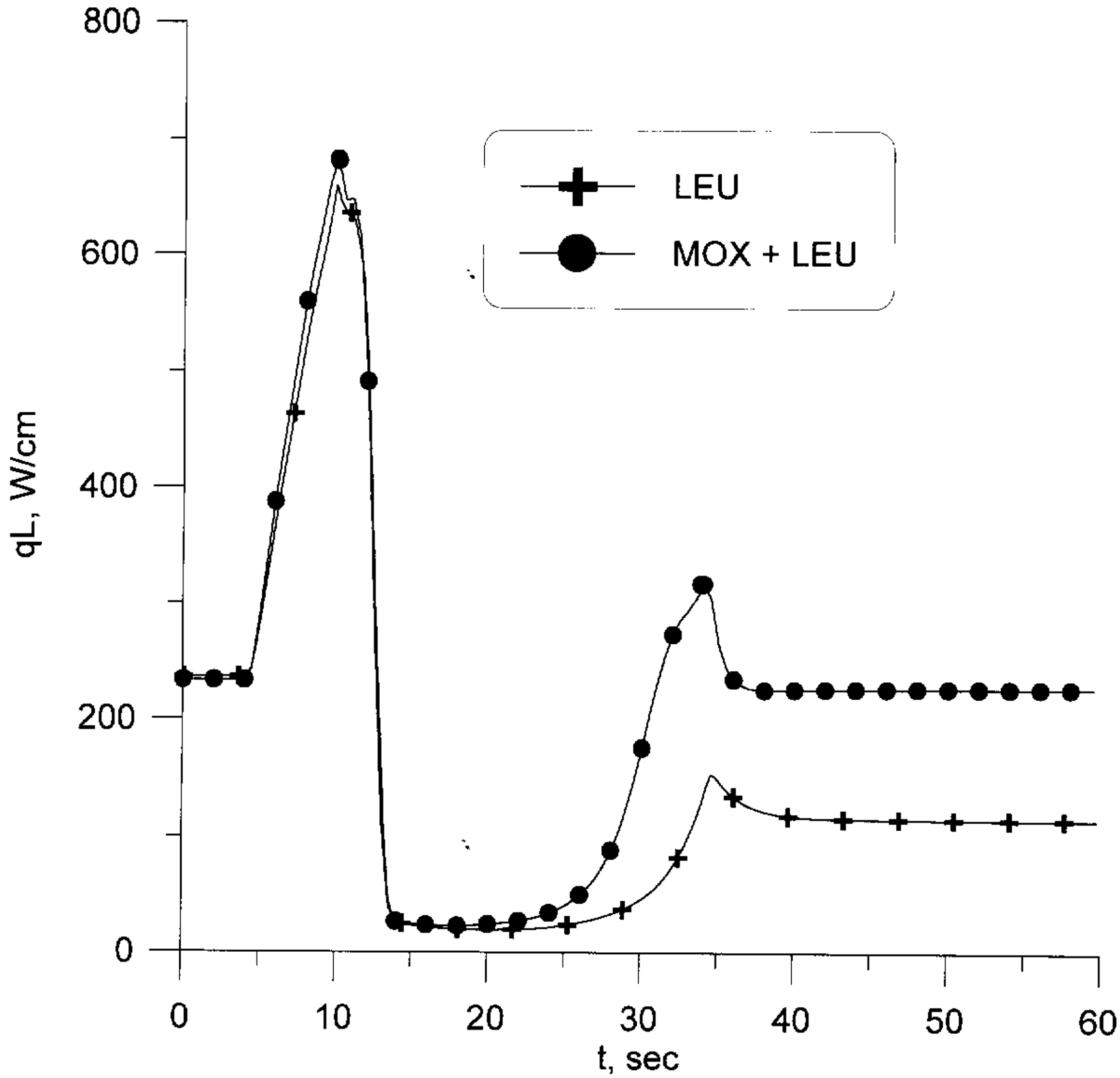


Fig. 6-5. Maximal Neutron Power. Loop Put into Operation & Boron Dilution

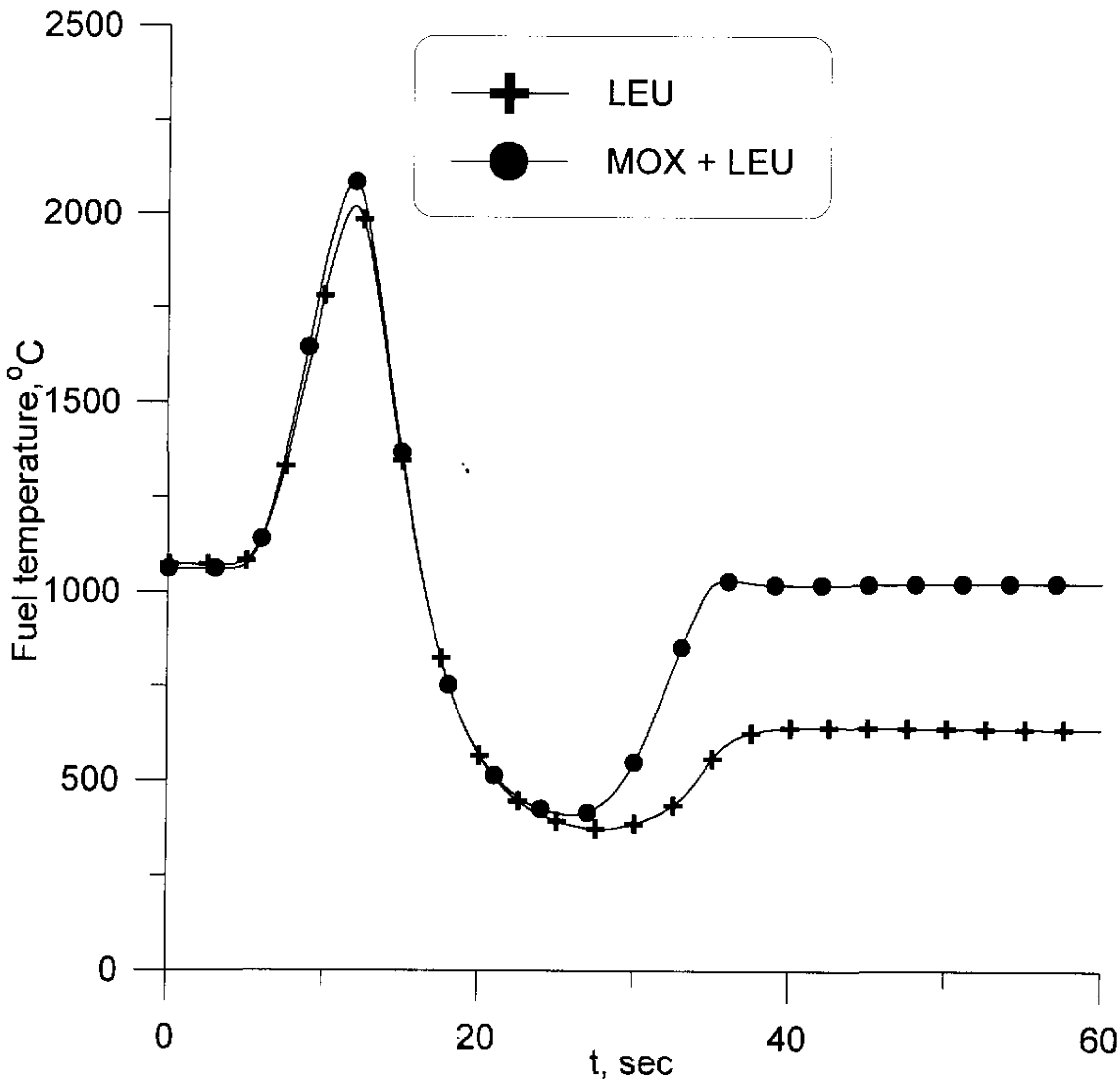


Fig. 6-6. Maximal Fuel Temperature. Loop Put into Operation & Boron Dilution

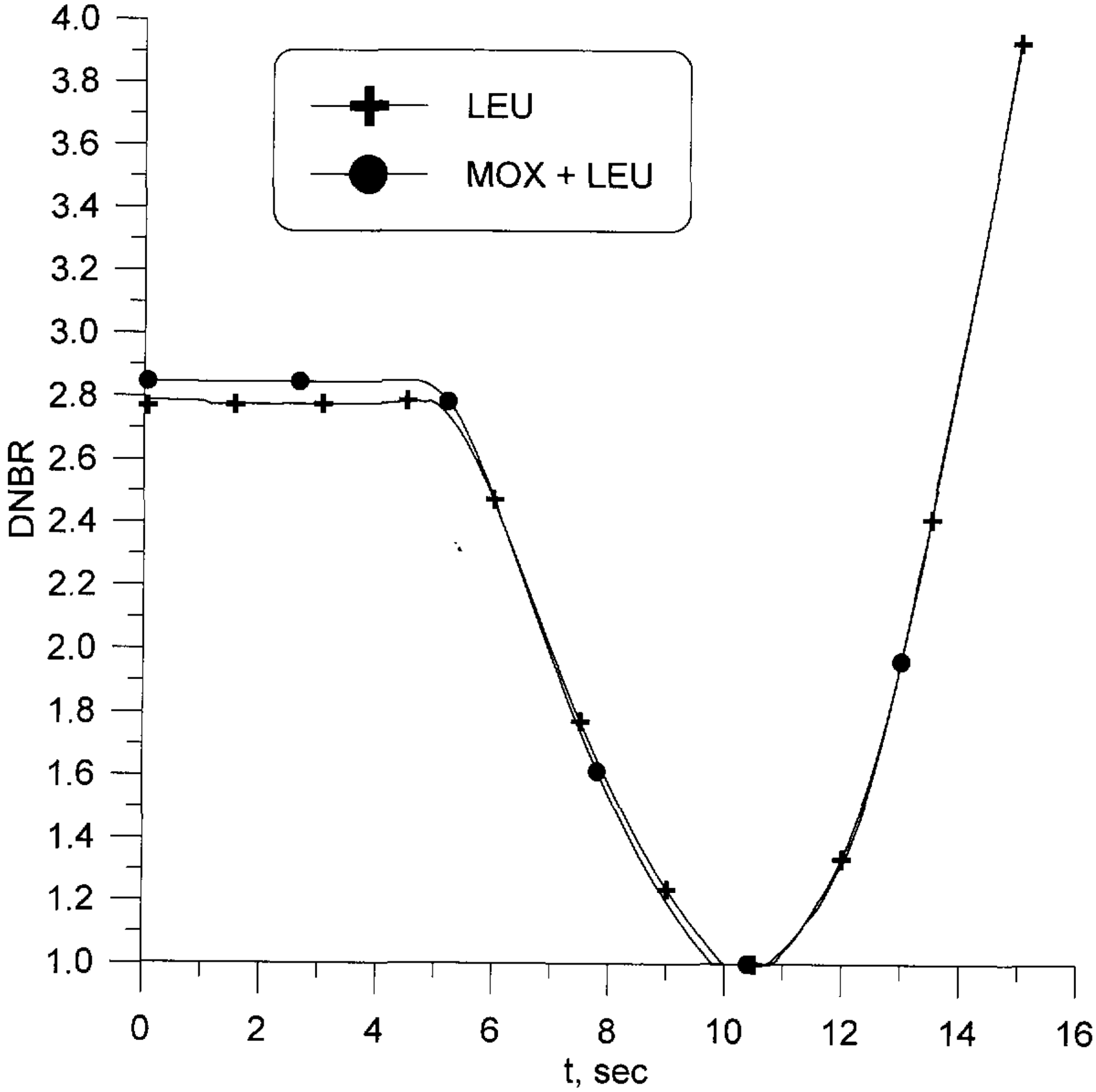


Fig. 6-7. Minimal DNBR. Loop Put into Operation & Boron Dilution

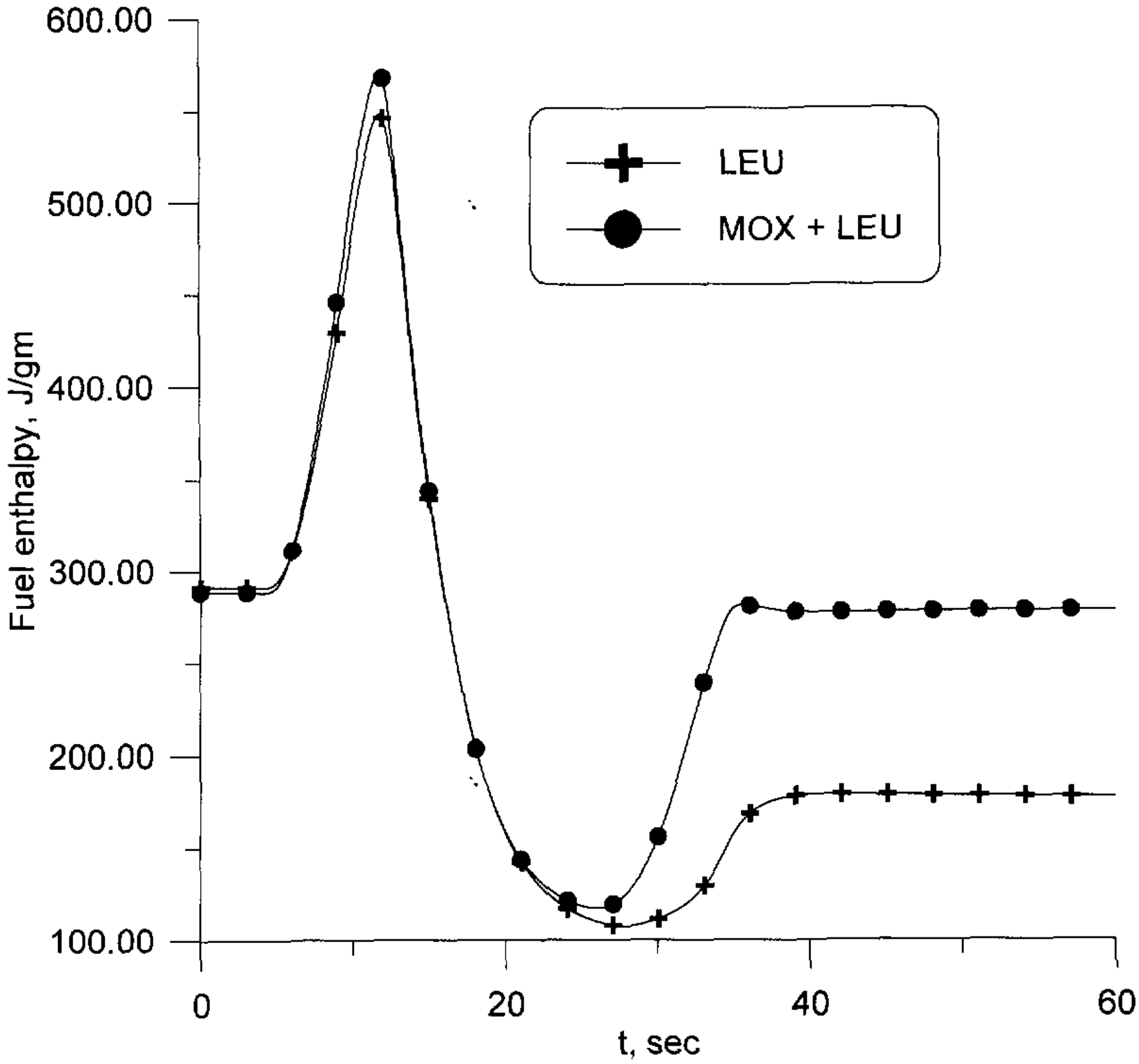


Fig. 6-8. Maximal Fuel Enthalpy. Loop Put into Operation & Boron Dilution

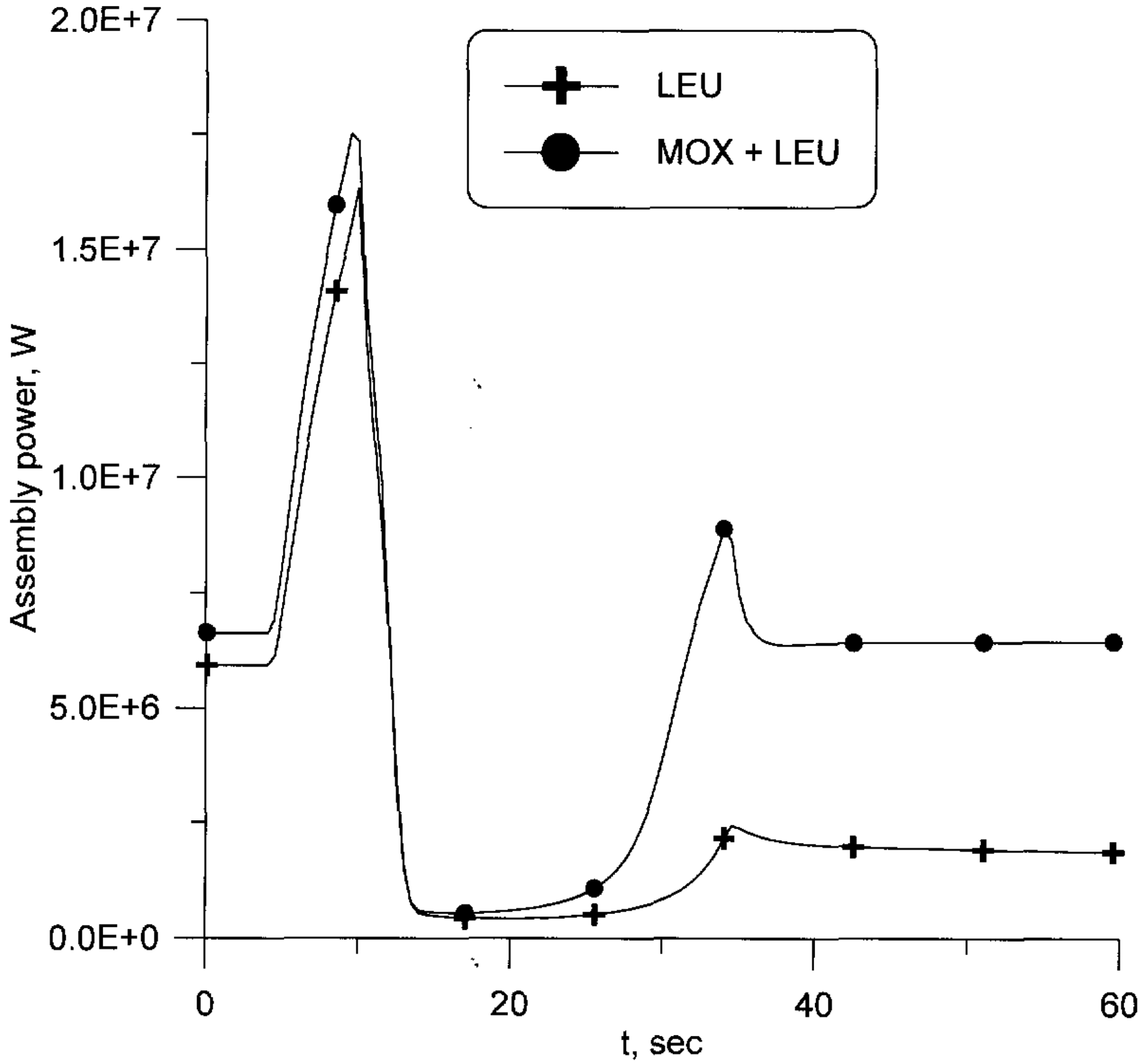


Fig. 6-9. Power of Assembly N 4. Loop Put into Operation & Boron Dilution

CONCLUSION

The analysis of results of the three dynamic problems solution allows to conclude, that their solution with the help of different codes will help to execute verification of codes, which one in further will be used for modelling of the transition and emergency modes of VVER operation.

These results obtained by 3D kinetics model of the Russian code NOSTRA can be used for comparison with a point kinetics model that is often used in accident analysis codes such as RELAP. In future the present analysis should be used for 3D model introduction in RELAP code.

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2. In-core fuel management code package validation for WWERs. IAEA-TECDOC-847. November 1995.

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