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Core Benchmarks Description Report

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Russian Research Center "Kurchatov Institute"

A Russian Contribution to the
Fissile Materials Disposition Program

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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

Core Benchmarks Description

Report

General Order 85B-99398V

Project Manager

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Moscow 2000

ACRONYMS

Russian		Western Equivalent
BOC	Beginning Of fuel Cycle	BOC
СВП или BPR	Burnable Poison Rod	BPR
DTC	Doppler Temperature Coefficient	DTC
EFPD	Effective Full Power Day	EFPD
EOC	End Of fuel Cycle	EOC
ТВС, FA	Fuel Assembly	FA
FP	Fission Products	FP
ТМ	Heavy Metal	HM
КИ	Kurchatov Institute	KI
LWR	Light Water Reactor	LWR
ОТВС	Leading Test Assembly	LTA
МКУ	Minimum Controllable reactor power Level	MCL
MDC	Moderator Density Coefficient	MDC
MOX	Mixed Oxide (uranium-plutonium fuel)	MOX
MTC	Moderator Temperature Coefficient	MTC
АЭС	Nuclear Power Plant	NPP
ОР	Regulatory Body (Control Rod)	CR
PWR	Pressurized-Water Reactor	PWR
СУЗ	Reactor Control and Protection System	RPS
твэг	Uranium-gadolinium fuel pin	tveg
UOX	Uranium Oxide Fuel	UOX
ВВЭР	Russian water-water reactor	VVER

Summary

The document issued according to Work Release KI-WR04RTP. P. 00-12-A describes the list of benchmarks and functionals necessary for verification of computer package of Russian Research Center "Kurchatov Institute": BIPR-7A, PERMAK-A and others. The first-step and future benchmarks are marked. The order of testing and the use of testing works results in certification documents are described.

Content

Introduction

Actual regulations while designing of new fuel cycles for nuclear power installations comprise a calculational justification to be performed by certified computer codes. It guaranties that obtained calculational results will be within the limits of declared uncertainties that are indicated in a certificate issued by Gosatomnadzor of Russian Federation (GAN) and concerning a corresponding computer code. A formal justification of declared uncertainties is the comparison of calculational results obtained by a commercial code with the results of experiments or of calculational tests that are calculated with an uncertainty defined by certified precision codes of MCU type or of other one. The actual level of international cooperation provides an enlarging of the bank of experimental and calculational benchmarks acceptable for a certification of commercial codes that are being used for a design of fuel loadings with MOX fuel. In particular, the work is practically finished on the forming of calculational benchmarks list for a certification of code TVS-M as applied to MOX fuel assembly calculations. The results on these activities are presented in [1] и [2].

The next step is a preparation of calculational benchmarks for certificating of the codes BIPR7-A and PERMAK-A with the constants libraries prepared by TVS-M. The above-mentioned codes are the base ones in RRC KI and are ordered for using at VVER NPPs while authorizing new fuel cycles for VVERs. The particularity of the actual working step is that the codes BIPR7-A and PERMAK-A are directed to calculations of rather wide spectrum of core neutronics characteristics with the dependence on fuel irradiation (see Annex A). Besides the code PERMAK-A is intended for neutronics field calculation on the border of a core and a reflector. The tests needed for commercial codes verification according to the material from Annex A can be conditionally subdivided into three groups:

1- Group of comparative tests directed to a comparison of results obtained by the codes of a similar type and demonstrating a whole fuel cycle calculation quality (beginning from some MOX FAs introduction into core until an equilibrium cycle with 30% MOX fuel fraction);

2- Group of tests based on the combination of potentials of commercial and precision codes. It is reasonable to apply this group to a solution of such tasks as 3-D neutron distributions and other functionals within the frame of real 3-D VVER-1000 core model.

3- Group of tests directed to precision calculations of different systems. Within the frame of this group the set of functionals is to verified if there is no corresponding experimental information for such functionals or if an access to this information is embarrassed. This tests group is directed to a verification of BIPR7-A and PERMAK-A calculational models taking into account a radial reflector configuration.

The tests subdividing into above-mentioned groups is defined first of all by a type of verification material (from the point of view of their "precisionness") that is to be used for a justification of calculational precision of different functionals. Obviously that a precision codes use is almost impossible in the 1-st tests group. So calculational uncertainties of such functionals as a cycle length etc. can be justified in the best case by a comparison of results issuing from different packages of similar type. In the second group a combination of precision and commercial codes is possible. And at last the third group of tests is intended for a creation of representative benchmarks using precision codes. Due to absence of experience in weapon-grade plutonium use in VVER core, a benchmarks representativity can be ensured only by involved precision codes and the last ones are supposed to be certified on the concerned tasks type using benchmarks experiments. Taking into account the demands being applied to BIPR7-A and PERMAK-A it seems reasonable to base the benchmarks to be proposed on calculations of "core with a reflector" type configurations with fuel burnup simulation (see Annex A). In general the tests to be proposed must reflect neutron distributions in core and on its border taking into account neutron leakage processes from a system etc. Neutron transfer modeling in such systems should be performed with a minimum of approaches.

Within the frame of the presented work it seems reasonable to consider the 3-d tests group. So in Annexes B and C the list of tests is presented that is intended for a testing of commercial codes package (BIPR7-A, PERMAK-A and others) using neutron few-group libraries prepared by TVS-M:

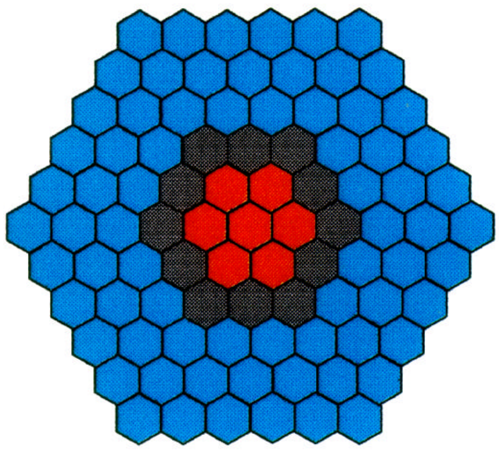
- in Annex B a set of systems is considered that contains 7 advanced VVER-1000 FAs surrounded by the steel buffer in the water reflector –«minicore»;
- in Annex C the practically real 2D model of VVER-1000 core is considered. The following loadings named "Core_2D-1000" are described: homogeneous – with MOX or UOX FAs, heterogeneous - 30% MOX core and a core with 3 MOX LTAs.

1. Description of the tests group "minicore"

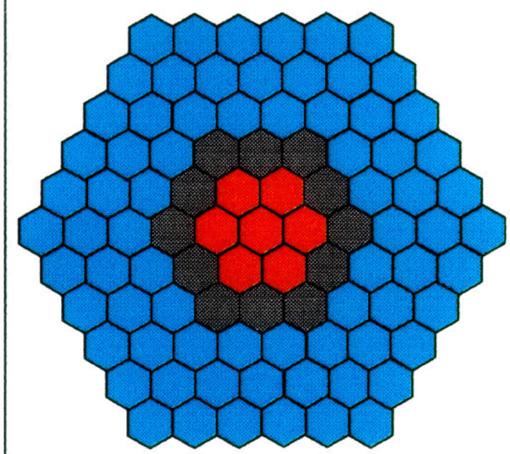
As it has been mentioned above this tests group is directed to the use of precision codes (MCU and others) allowing a detailed modeling of neutron transfer in the systems of any geometry. The size of the modeled system and the configuration of FA have been chosen taking into account a real potential of precision codes to ensure cell-by-cell registration of different functionals with a reasonable computer time consumption. The detailed description of the considered systems is presented in Annex **B** (see Fig. B-7 and Table B-4). This set comprises 7 different core configurations.

The detailed geometry description of the considered systems and fuel assemblies types is given in Tables B-3 and B-4. Below the brief commentaries to the considered systems are presented:

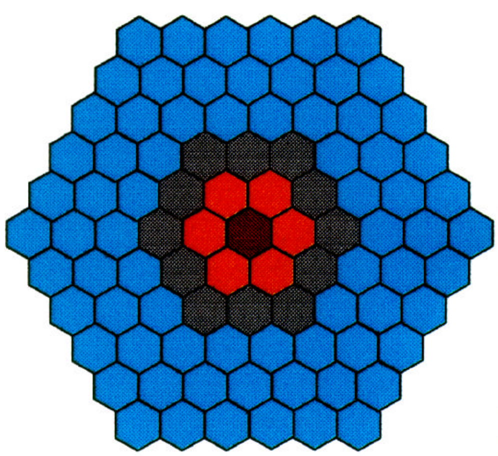
1. Core with 7 equivalent non-graded UOX fuel assemblies (U-235 enrichment is 3.7%)



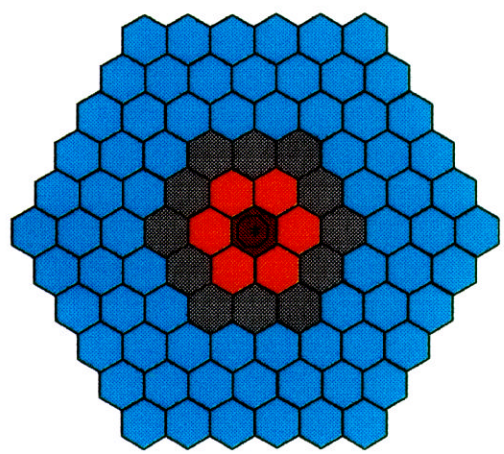
5. Core with 7 equivalent non-graded MOX fuel assemblies (Pu enrichment is 3.5%)



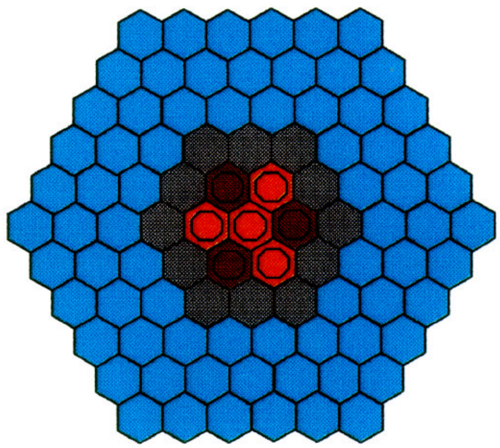
2. Core with 6 equivalent non-graded UOX fuel assemblies (U-235 enrichment is 3.7%) with the non-graded Pu FA in the center (Pu enrichment is 3.5%)



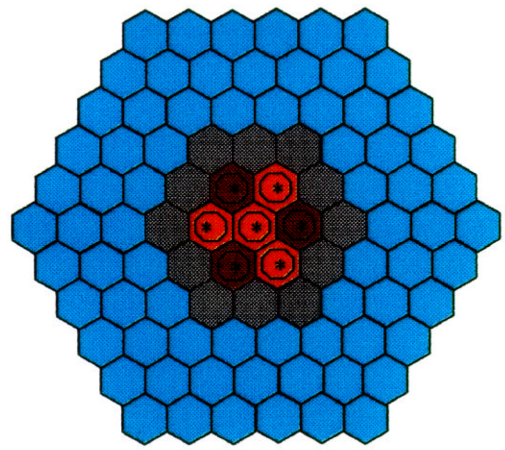
4. Core with 6 equivalent non-graded UOX fuel assemblies (U-235 enrichment is 3.7%) with the graded Pu and "tvegged" FA in the center



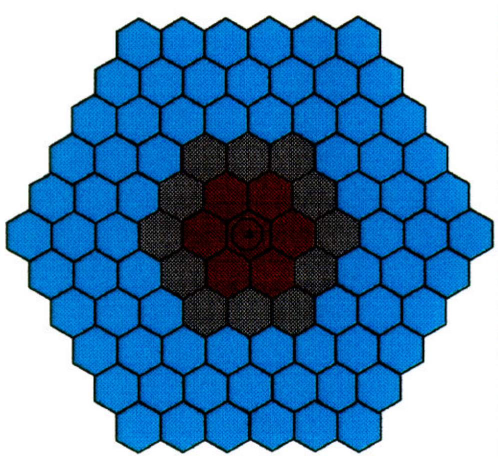
3. Core with 4 equivalent graded UOX fuel assemblies and 3 equivalent graded MOX fuel assemblies.



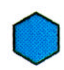
7. Core with 4 equivalent graded and "tvegged" UOX fuel assemblies and 3 equivalent graded and "tvegged" MOX fuel assemblies.



6. Core with 6 equivalent non-graded MOX fuel assemblies (Pu enrichment is 3.5%) with the graded and "tvegged" Pu FA in the center



Legend

	Water reflector assembly
	Steel buffer assembly
	Non-graded fuel assemblies with UOX or MOX
	Graded fuel assemblies with UOX or MOX
	"Tvegged" and graded fuel assemblies with UOX or MOX

This group of tests is the most acceptable for a verification of PERMAK-A that is intended for duffusion fine-mesh calculations of reactor core with a reflector. The general structure and the list of testing tasks allow to justify on an acceptable level a PERMAK-A use in MOX-fuel core designing. It seems perspective within the frame of this tests group, for the 4-th system, to model a pin-by-pin fuel burnup process using the precision codes potential. It could allow more verification of PERMAK-A on the base of burnup-type tasks. The use of this tests group for BIPR7-A is principally possible. But a comparative analysis will be valuable only in the case of acceptable results coherence because the proposed core configuration does not correspond to the region of validity of the BIPR7-A option that is to be certified for MOX-fuel cores.

1.1. States description

As it was mentioned above the objective of this document was to formulate calculational benchmarks for a certification of the commercial codes directly used in reactor design and operation. Annex A describes possibilities and fields of applications of corresponding codes. Being based on this information it is reasonable to consider the following set of core states:

1. Full power, non-poisoned (without Xe and Sm), with boron in coolant;
2. Hot, non-poisoned, zero power with boron in coolant;
3. Hot, non-poisoned, zero power without boron in coolant;
4. Accidental without boron in coolant;
5. Cold, non-poisoned, with boron in coolant;
6. Cold, non-poisoned, without boron in coolant;
7. Full power, non-poisoned, with the absorbants of natural boron;
8. Full power, non-poisoned, with the absorbants of enriched boron;
9. Full power, non-poisoned, with boron BPRs;

The detailed parameters of the above-mentioned states are presented in Annex B (see Table B-5)

1.2. Burnup procedure description

Fuel burnup is simulated only for the System N 4 in state 1 taking into account Xe and Sm equilibrium poisoning (see Tables B-4 and B-5).

Average specific power is of 108 MW/м^3 (Averaging is performed over the FAs N 1-7, Fig. B-7). The range of average fuel burnup over the mentioned region is 0-24 MW-day/kg HM. For the "tvegged" FAs a nuclide composition is to be calculated with the step 0.1 MW-day/kg HM in the burnup interval 0-10 MW-day/kg HM and with the step 1 MW-day/kg HM in the burnup interval 10-24 MW-day/kg HM (everywhere burnup values averaged over a system are mentioned).

Precision codes calculate a nuclide composition evolution in average over every pin in a system. While modeling a nuclide composition evolution in tvegs the last will be subdivided into five concentric zones of equal volume. A further evolution of a nuclide composition is to be modeled in average over a concentric zone.

1.3. Functionals to Be Registered

Below the general list of functionals to be registered is presented. Within the frame of actual investigations it is supposed that every system (see Table B-4) will be calculated in the states 1-9 (Table B-5). The value of border energy of the 1-th and 2-nd group is equal to 0.625ev.

1.For every system as a whole (Fig.B-6, B-7):

1.1. Effective multiplication factor;

1.2. Ko for the region limited by the FAs N 1-7 (Fig.B-7);

1.3. Effective fraction of delayed neutrons;

1.4. Life-time of prompt neutrons;

1.5. Neutron absorption rates in fuel assemblies (FAs N 1-7), in the buffer (FAs N 8-19), in the reflector (FAs N 20-91), leakage from the system and integral one-group flux in the mentioned regions.

Remark:

For the system 4 the mentioned functionals are to be presented for the following burnup values: 0, 12, 24 MW-day/kg HM. Besides in state 1 (taking into account Xe and Sm equilibrium poisoning) the functionals 1.1, 1.2 and 1.5 are to be calculated versus average system fuel burnup.

2. In every fuel assembly of a system (Fig. B-6):

2.1. Powers (in FAs N 1-7), one-group neutron reaction rates (fission, absorption and generation), Ko, neutron migration squares, effective fractions of delayed neutrons, one-group effective cross-sections of neutron fission, absorption and generation, burnups averaged over a FA ;

2.2. Neutron fluxes, neutron reaction rates (fission, absorption and generation) - in two groups, two-groups effective cross-sections of neutron fission, absorption and generation.

Remark:

If a code can register (calculate) neutron migration squares and effective fractions of delayed neutrons for FAs it is desirable to briefly describe a calculational methodology.

For the system 4 the mentioned functionals are to be presented for the following burnup values: 0, 12, 24 MW-day/kg HM (it means an average burnup over a system). Besides in state 1 (taking into account Xe and Sm equilibrium poisoning) the functionals 2.1 are to be calculated versus average FA fuel burnup.

3. In the register areas of every FA (Fig. B-5, B-6 and B-8):

3.1. In one-group approach – average burnup, power (FAs N 1-7), neutron fluxes, neutron reaction rates (fission, absorption and generation), Ko, neutron migration squares, effective fractions of delayed neutrons, effective cross-sections of neutron fission, absorption and generation;

3.2. In two-groups approach - neutron fluxes, neutron reaction rates (fission, absorption and generation), effective cross-sections of neutron fission, absorption and generation.

Remark:

For the system 4 the mentioned functionals are to be presented for the following burnup values: 0, 12, 24 MW-day/kg HM (it means an average burnup over a system). Besides in state 1 (taking into account Xe and Sm equilibrium poisoning) the functionals 3.1 are to be calculated versus average system fuel burnup.

4. In the same inter-assembly cell types for every register area in FAs N 1-91 (see.Fig. B-1 ÷ B-8 and Table.B-3):

4.1. In one-group approach – average fuel burnup for every type, average power, Ko, neutron reaction rates of fission, absorption and generation, neutron flux, effective cross-sections of neutron fission, absorption and generation.

4.2. In two-group approach – neutron reaction rates of fission, absorption and generation, neutron flux, effective cross-sections of neutron fission, absorption and generation.

Remark:

For the system 4 the mentioned functionals are to be presented for the following burnup values: 0, 12, 24 MW-day/kg HM (it means an average burnup over a system). Besides in state 1 (taking into account Xe and Sm equilibrium poisoning) the functionals 4.1 are to be calculated versus average system fuel burnup.

The strict definition of a cell type is presented in Annex B. In a simplified interpretation two cells correspond to the same type if a first one can be completely represented by a second one by using a removal along the axes "x" and "y". So there are 6 types of "gap" cells and 6 types of "corner" cells in every FA. For the 4-th system cell types are defined for a zero burnup.

5. In every inter-assembly cell (see Fig. B-5):

5.1. In one-group approach – average burnup, power, Ko, neutron reaction rates of fission, absorption and generation, neutron flux, effective cross-sections of neutron fission, absorption and generation.

5.2. In two-groups approach – neutron reaction rates of fission, absorption and generation, effective cross-sections of neutron fission, absorption and generation.

Remark:

For the system 4 the mentioned functionals are to be presented for the following burnup values: 0, 12, 24 MW-day/kg HM (it means an average burnup over a system). Besides in state 1 (taking into account Xe and Sm equilibrium poisoning) the functionals 5.1 are to be calculated versus average system fuel burnup.

2. Description of the tests group "Core_2D-1000"

As it has been mentioned in Chapter 1, actually a use of precision codes for a direct modeling of equilibrium reloading process is practically impossible. At the same time a calculation of some VVER-1000 core state with a defined isotopic composition is a realizable task for the actual level of precision codes development if we do not detail isotopic compositions of every fuel pin in a system. So it seems perspective to develop a tests group that will combine a simplified fuel burnup model and a precision neutronics calculation of a whole core. Such tests group will allow to verify commercial codes BIPR7-A and PERMAK-A by a direct comparison with precision codes for a full-scale VVER-1000 core calculation. Annex C (Table C-1) describes 5 VVER-1000 core compositions. Within the frame of actual work it seems reasonable to consider only 2D model of reactor core. The detailed description of the tested cases is presented in Annexes B and C. Below a short explanation of every case is presented:

1. Homogeneous core (Fig.C-7), formed from fresh UOX FAs of 3.7% enrichment (Table B-3, fuel assembly K1);
2. Homogeneous core (Fig.C-7), formed from fresh Pu FAs of 3.6% enrichment (Table B-3, fuel assembly K2);
3. Equilibrium core loading with 30% MOX fuel (BOC). Fuel loading pattern is presented in Fig. C-8 (type 1 – UOX FAs, type 2 – MOX FAs);
4. Equilibrium core loading with 30% MOX fuel (EOC). Fuel loading pattern is presented in Fig. C-9 (type 1 – UOX FAs, type 2 – MOX FAs);
5. Core loading with 3 LTAs (type 2, Fig. C-11).

2.1 States description

Within the frame of the actual work statement the following core states are proposed to be considered (see Table C-2):

- Full power with boron in coolant – B1;
  Full power without boron in coolant – E1;
- Hot state with boron in coolant – B2;
  Hot state without boron in coolant – E2;
- Cold state (temperature is 27°C) with boron (2800 pcm) in coolant – B3;
- State at MCL with all CRs inserted into the core – E3;
- State at MCL with all CRs inserted into the core except of one (**, Fig. C-10) – E4.

In detail the parameters of different materials for different core states are presented in Table C-2.

2.2. Burnup procedure description

The proposed in Annex C test fuel loadings of VVER-1000 type are based on 30% MOX core investigations /3/ or on a loading with 3 MOX LTAs /4/. Earlier it has been mentioned that the objective of the actual work is a verification of the codes BIPR7-A and PERMAK-A by comparing with calculations to be obtained by precision codes. So a real VVER-1000 core structure has been simplified in order to consider only two types of fuel assemblies with a fixed set of burnups. It is presented in Annex C that UOX FAs (type 1) have the average fuel burnup of 0, 15, 32, 40, 48 MW-day/kg HM and MOX FAs (type 2) - 0, 17, 33, 41 MW-day/kg HM. It was assumed that fuel pins of the same type have the same nuclide composition being a function of average FA fuel burnup. So a fuel burnup process has been simulated by TVS-M for UOX and MOX FAs in order to obtain nuclide compositions of different materials. The nuclide compositions obtained for different fuel pins of a same type have been averaged and have been presented as new materials. For example the material U_4.2:15 corresponds to the averaged nuclide composition of the material U_4.2 in the corresponding FA with the average burnup of 15 MW-day/kg HM. This approach permits to use the formal FA description presented in Table B-3 with the correction on an average FA burnup (with the correction on a material subtype, see Fig. C-8, C-9, C-11). In other words in order to receive material name for the patterns C-8, C-9 and C-11 it is needed to add ":material subtype" to a material name from Table B-3. Isotopic compositions with the renewed material name is presented in Table B-1.

2.3. Functionals to be registered

After having performed corresponding calculations depending on code possibilities the following functionals are to be formed:

1. In the whole system (Fig. B-1 and B-2):
   1.1. Effective multiplication factor;
   1.2. Ko for the object L-1000 (Fig. C-5);
   1.3. Effective fraction of delayed neutrons in every FA and for the whole system;
   1.4. Life-time of prompt neutrons;
   1.5. Absorption fraction (per one born neutron) in the fuel assembly grid -"L1000", in the water gap - "W_gap", in the cooling tubes of the buffer –"HoleV", in the buffer itself– "V", in the steel reactor barrel– "C3", downstream part – "C2", steel vessel – "C1", leakage from the system and integral one-group flux in the mentioned regions.

2. In every element of the register area (Fig. C-4 – C-6)
   2.1. Average burnup, power, Ko, neutron migration square, effective fraction of delayed neutrons, in one-group approach: neutron flux,

neutron reaction rates of fission, absorption and generation; effective cross-sections of neutron fission, absorption and generation;

2.2. In two-groups approach: neutron reaction rates (fission, absorption and generation), neutron flux, effective cross-sections of neutron fission, absorption and generation.

3. In the register areas for every element of register set (Fig.B-8) :

3.1. In one-group approach – power, Ko, neutron reaction rates of fission, absorption and generation, neutron flux, effective cross-sections of neutron fission, absorption and generation are to be registered.

3.2. In two-group approach – neutron reaction rates of fission, absorption and generation, neutron flux, effective cross-sections of neutron fission, absorption and generation in FAs are to be registered.

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2. A.P. Lazarenko, et al.,. "Benchmark Calculations For VVER-1000 Fuel Assemblies Using Uranium or MOX Fuel." ANS International Topical Meeting on Advances in Reactor Physics and Mathematics and Computation into the Next Millennium. May 7-12, 2000, Westin William Penn Hotel Pittsburgh, Pennsylvania, USA. Published by the American Nuclear Society, Inc. 555 North Kensington Avenue, La Grange Park, IL 60526, USA. ISBN: 0-89448-655-1; ANS Order No. 2700281; Copyright © 2000 ANS.

3. A.M.Pavlovichev, Y.A.Styrine, S.N.Bolshagin, Documentation of core configuration for equilibrium MOX core with U-Gd fuel pins. RRC KI, Moscow 2000.

4. A.M.Pavlovichev. MOX LTA design. RRC KI, Moscow, 24-28 July 2000.

Annex A

1. Brief characteristics of the codes to be tested

1.1 Code TVS-M

1. Destination and application area:

Application area of the code – calculations of neutronics characteristics of homogeneous fuel grids and LWR fuel assemblies. Main code destination – preparation of few-groups neutronics constants for LWR core calculations, primarily – for VVERs.

2. Verified range of parameters values:

Volume water/fuel ratio 0.5-3.5
Water density in a fuel grid or in a FA 0.2-1.0 g/cm^3
Water temperature until 300^{0}C
Fuel temperature until 750^{0}C
Initial content of fissile material in a fuel (%) 1.6-6.0
Fuel burnup until 60 MW-day/kg HM
Fuel pellet diameter 0.6-1.2 см

3. Information on neutronics constants libraries of the code:

The library of nuclear data obtained on the base of the constants library DLC/MCUDAT-1.0 of the code MCU-RFFI is used. Neutron fission issues were obtained on the base of ENDF-B/VI library.

4. The functionals to be verified in the allowed parameters values with the constants prepared by the code TVS-M:

- K_{eff} , Ko for homogeneous grids with fresh UOX fuel for cold, hot and full-power states.
- K_{eff} , Ko for fuel assemblies with fresh and spent UOX fuel including FAs with CRs, (boron) BPRs and tvegs for cold, hot and full-power states.
- K_{eff} , Ko for homogeneous grids with fresh MOX fuel for cold, hot and full-power states.
- K_{eff} , Ko for fuel assemblies with fresh and spent MOX fuel including FAs with CRs, (boron) BPRs and tvegs for cold, hot and full-power states.
- Reactivity effects for UOX and MOX fuel on fuel temperature, boron concentration, coolant density, xenon concentration etc.
- Effective fraction of delayed neutrons for UOX and MOX FAs.
- Precisions of power calculations in a fuel pin for different fuel burnups.
- Precisions of calculations of neutron reaction rates of fission, absorption and generation in a fuel pin for different fuel burnups.
- Precisions of calculations of different nuclides contents in UOX and MOX fuel for different fuel burnups.

1.2 Code BIPR7-A

1. Destination and application area:

1.1. Destination

- Calculations of criticality parameters, reactivity effects and coefficients, CRs efficiency, core power distributions;
- Calculational modeling of fuel burnup and reloading, Xenon-135 and Sm-149 transient processes for VVER fuel loadings;
- Automatic calculation in combination with the code PERMAK-A of neutronics core characteristics according to the regulating document "List of operation neutronics calculations and experiments for VVER-440 (VVER-1000) fuel loadings".

1.2. Reactor type

Light-water with the forced coolant circulation (VVER type).

1.3. Regimes

Core states realizable both in normal operation and in operational occurrences.

1.4. Limits on an application

Calculations of core neutronics characteristics of VVER with uranium and uranium-gadolinium fuel.

1.5. Allowable parameters values

- Water/uranium ratio 1.5 –2.5;
- Initial fuel enrichment 1.6 – 4.4%;
- Fuel pin outer diameter 0.8 –1.2 cm;
- Boron concentration in a coolant 0 – 4000 ppm;
- Fuel burnup 0 –60000 MW-day/ т HM;
- Coolant density 0.6 –1.0 g/cm^3;
- Calculational node size (FA width across flats or an axial node size) h must meet the requirement h >>L, where L – thermal neutrons diffusion length in a fuel pins grid;
- For all the FAs in a core the requirement must be met: K*h/2 < 0.8, where K^2=(Kinf\ K_{eff}-1)/M^2 , h – calculational node size (FA width across flats or an axial node size).

1.6. The functionals to be verified in the allowed range of parameters values with the constants prepared by the TVS-M code:

- Critical boron concentration in a coolant during fuel loading operation (full-power state);
- CRs efficiency as a whole;
- Single group CRs efficiency;
- Temperature coefficient in BOC;

- Assembly-by-assembly power peaking factor;
- Node-by-node power peaking factor;
- Power reactivity coefficient (full-power state);
- Critical temperature of a shut-down reactor in EOC (C_{H3BO3}=0);
- Fuel cycle length (life-time of a fuel loading).

Calculational precisions of the mentioned functionals must be justified by a comparison of calculational and experimental data obtained during VVER-1000 start-up and operation (for two-, three- and four-years fuel cycles).

2. Information on the data-base (neutron constants libraries) to be ised in a code:

Constants for BIPR7-A are to be prepared by the code TVS-M.

1.3 Code PERMAK-A

1. Destination and application area:

1.1. Destination

Multi-layer 2D calculations of core neutronics characteristics according to the regulating document "List of operation neutronics calculations and experiments for VVER-440 (VVER-1000) fuel loadings".

1.2. Reactor type

Light-water with the the forced coolant circulation (VVER type).

1.3. Core states realizable both in normal operation and in operational occurrences.

1.4. Limits on an application

Calculations of core neutronics characteristics of VVER with uranium and uranium-gadolinium fuel.

1.5 Allowable parameters values

- Water/uranium ratio 1.5 –2.5;
- Initial fuel enrichment 1.6 – 4.4%;
- Fuel pin outer diameter 0.8 –1.2 cm;
- Boron concentration in a coolant 0 – 4000 ppm;
- Fuel burnup 0 –60000 MW-day/ т HM;
- Coolant density 0.6 –1.0 g/cm^3;
- Linear fuel pin power 0-500 W/cm.

1.6 The functionals to be verified in the allowed range of parameters values with the constants prepared by the TVS-M code:

- Effective multiplication factor;

- Assembly-by-assembly power peaking factor in a layer;
- Pin-by-pin power peaking factor in a FA;
- Relative power of fuel pins close to the water holes;
- Relative power of fuel pins close to the absorbing pins;
- Relative power of fuel pins close to the uranium-gadolinium pins (tvegs).

2. Information on the data-base (neutron constants libraries) to be used in a code:

Constants for BIPR7-A are to be prepared by the code TVS-M.

3. Specific conditions:
   - Calculations are to be performed in the combination with BIPR7-A;
   - Calculations are to be performed only for the serial Fa geometry.

Annex B

Table. B- 1 Materials description.

Material Name	Comments	Material isotopic composition (isotope, nuclear concentration [10^{-24} cm^{-3}])	
General material type: Fuel			
U_3.7 U_3.7:0	UOX fuel with U ^{235} enrichment 3.7% wt.	^{235}U ^{238}U ^{16}O	7.9649E-04 2.0469E-02 4.2530E-02
U_4.2 U_4.2:0	UOX fuel with U ^{235} enrichment 4.2% wt.	^{235}U ^{238}U ^{16}O	9.0411E-04 2.0362E-02 4.2532E-02
PU_2.4 PU_2.4:0	MOX fuel with fissile plutonium isotopes enrichment 2.42% wt.	^{235}U ^{238}U ^{16}O ^{238}Pu ^{239}Pu ^{240}Pu ^{241}Pu ^{242}Pu	4.3057E-05 2.0660E-02 4.2508E-02 7.2271E-07 5.0579E-04 3.5961E-05 6.4023E-06 2.3413E-06
PU_2.7 PU_2.7:0	MOX fuel with fissile plutonium isotopes enrichment 2.69% wt.	^{235}U ^{238}U ^{16}O ^{238}Pu ^{239}Pu ^{240}Pu ^{241}Pu ^{242}Pu	4.3057E-05 2.0598E-02 4.2508E-02 8.0774E-07 5.6222E-04 3.9987E-05 7.1160E-06 2.6131E-06
PU_3.6 PU_3.6:0	MOX fuel with fissile plutonium isotopes enrichment 3.62% wt.	^{235}U ^{238}U ^{16}O ^{238}Pu ^{239}Pu ^{240}Pu ^{241}Pu ^{242}Pu	4.3057E-05 2.0386E-02 4.2506E-02 1.0841E-06 7.5661E-04 5.3794E-05 9.5720E-06 3.5119E-06

Table B-1 (continued)

Material Name	Comments	Material isotopic composition (isotope, nuclear concentration [10^{-24} cm^{-3}])	
U_4.2:15	Uranium fuel with U ^{235} enrichment 4.2% Wt. corresponding to FA average burnup 15 MW-day/kg HM.	^{235}U	5.8139E-04
		^{236}U	5.7700E-05
		^{238}U	2.0161E-02
		^{237}Np	3.7658E-06
		^{238}Pu	4.5135E-07
		^{239}Pu	9.6584E-05
		^{240}Pu	1.7820E-05
		^{241}Pu	7.8291E-06
		^{242}Pu	8.5918E-07
		^{241}Am	9.6169E-08
		^{16}O	4.2532E-02
		^{135}Xe	9.7806E-09
		^{149}Sm	9.1807E-08
		^{151}Sm	3.8524E-07
		^{99}Tc	1.9865E-05
		^{103}Rh	1.1241E-05
		^{131}Xe	9.0873E-06
		^{133}Cs	2.1660E-05
		^{143}Nd	1.7145E-05
		^{145}Nd	1.2231E-05
		^{147}Pm	5.1135E-06
		^{152}Sm	2.0001E-06
U_4.2:32	Uranium fuel with U ^{235} enrichment 4.2% Wt. corresponding to FA average burnup 32 MW-day/kg HM.	^{235}U	3.2990E-04
		^{236}U	9.7452E-05
		^{238}U	1.9905E-02
		^{237}Np	1.0351E-05
		^{238}Pu	2.7546E-06
		^{239}Pu	1.2788E-04
		^{240}Pu	4.4214E-05
		^{241}Pu	2.4361E-05
		^{242}Pu	6.9401E-06
		^{241}Am	5.7445E-07
		^{16}O	4.2532E-02
		^{135}Xe	8.6151E-09
		^{149}Sm	8.9565E-08
		^{151}Sm	4.9458E-07
		^{99}Tc	4.0006E-05
		^{103}Rh	2.2541E-05
		^{131}Xe	1.6792E-05
		^{133}Cs	4.3134E-05
		^{143}Nd	3.0721E-05
		^{145}Nd	2.3856E-05
		^{147}Pm	7.0579E-06
		^{152}Sm	4.0165E-06

Table B-1 (continued)

Material Name	Comments	Material isotopic composition (isotope, nuclear concentration [10^{-24} cm^{-3}])	
U_4.2:40	Uranium fuel with U ^{235} enrichment 4.2% Wt. corresponding to FA average burnup 40 MW-day/kg HM.	^{235}U	2.4314E-04
		^{236}U	1.0890E-04
		^{238}U	1.9773E-02
		^{237}Np	1.3577E-05
		^{238}Pu	4.6336E-06
		^{239}Pu	1.3134E-04
		^{240}Pu	5.4717E-05
		^{241}Pu	3.0501E-05
		^{242}Pu	1.1811E-05
		^{241}Am	8.2251E-07
		^{16}O	4.2532E-02
		^{135}Xe	7.9972E-09
		^{149}Sm	8.5421E-08
		^{151}Sm	5.3181E-07
		^{99}Tc	4.8545E-05
		^{103}Rh	2.6925E-05
		^{131}Xe	1.9402E-05
		^{133}Cs	5.2008E-05
		^{143}Nd	3.5092E-05
		^{145}Nd	2.8579E-05
		^{147}Pm	7.2585E-06
		^{152}Sm	4.7868E-06
U_4.2:48	Uranium fuel with U ^{235} enrichment 4.2% Wt. corresponding to FA average burnup 48 MW-day/kg HM.	^{235}U	1.7403E-04
		^{236}U	1.1634E-04
		^{238}U	1.9631E-02
		^{237}Np	1.6611E-05
		^{238}Pu	6.9525E-06
		^{239}Pu	1.3150E-04
		^{240}Pu	6.3380E-05
		^{241}Pu	3.5073E-05
		^{242}Pu	1.7650E-05
		^{241}Am	1.0198E-06
		^{16}O	4.2532E-02
		^{135}Xe	7.4105E-09
		^{149}Sm	8.0833E-08
		^{151}Sm	5.6226E-07
		^{99}Tc	5.6439E-05
		^{103}Rh	3.0621E-05
		^{131}Xe	2.1386E-05
		^{133}Cs	6.0042E-05
		^{143}Nd	3.8217E-05
		^{145}Nd	3.2830E-05
		^{147}Pm	7.2175E-06
		^{152}Sm	5.4601E-06

Table B-1 (continued)

Material Name	Comments	Material isotopic composition (isotope, nuclear concentration [10^{-24} cm^{-3}])	
U_3.7:15	Uranium fuel with U ^{235} enrichment 3.7% Wt. corresponding to FA average burnup 15 MW-day/kg HM.	^{235}U	4.8884E-04
		^{236}U	5.4042E-05
		^{238}U	2.0262E-02
		^{237}Np	3.7015E-06
		^{238}Pu	4.6522E-07
		^{239}Pu	9.5675E-05
		^{240}Pu	1.9149E-05
		^{241}Pu	8.3590E-06
		^{242}Pu	1.0147E-06
		^{241}Am	1.0325E-07
		^{16}O	4.2530E-02
		^{135}Xe	8.7774E-09
		^{149}Sm	8.3005E-08
		^{151}Sm	3.4935E-07
		^{99}Tc	1.9485E-05
		^{103}Rh	1.1189E-05
		^{131}Xe	8.9232E-06
		^{133}Cs	2.1245E-05
		^{143}Nd	1.6565E-05
		^{145}Nd	1.1935E-05
		^{147}Pm	4.9515E-06
		^{152}Sm	2.0058E-06
U_3.7:32	Uranium fuel with U ^{235} enrichment 3.7% Wt. corresponding to FA average burnup 32 MW-day/kg HM.	^{235}U	2.6496E-04
		^{236}U	8.8494E-05
		^{238}U	2.0000E-02
		^{237}Np	9.9287E-06
		^{238}Pu	2.7516E-06
		^{239}Pu	1.2378E-04
		^{240}Pu	4.5926E-05
		^{241}Pu	2.4695E-05
		^{242}Pu	7.7274E-06
		^{241}Am	5.7739E-07
		^{16}O	4.2530E-02
		^{135}Xe	7.7735E-09
		^{149}Sm	8.2706E-08
		^{151}Sm	4.5427E-07
		^{99}Tc	3.8798E-05
		^{103}Rh	2.2235E-05
		^{131}Xe	1.6280E-05
		^{133}Cs	4.1820E-05
		^{143}Nd	2.9007E-05
		^{145}Nd	2.2961E-05
		^{147}Pm	6.7031E-06
		^{152}Sm	3.9584E-06

Table B-1 (continued)

Material Name	Comments	Material isotopic composition (isotope, nuclear concentration [10^{-24} cm^{-3}])	
U_3.7:40	Uranium fuel with U ^{235} enrichment 3.7% Wt. corresponding to FA average burnup 40 MW-day/kg HM.	^{235}U	1.9092E-04
		^{236}U	9.7726E-05
		^{238}U	1.9864E-02
		^{237}Np	1.2901E-05
		^{238}Pu	4.5708E-06
		^{239}Pu	1.2659E-04
		^{240}Pu	5.6139E-05
		^{241}Pu	3.0494E-05
		^{242}Pu	1.2925E-05
		^{241}Am	8.0933E-07
		^{16}O	4.2530E-02
		^{135}Xe	7.2565E-09
		^{149}Sm	7.9571E-08
		^{151}Sm	4.9115E-07
		^{99}Tc	4.6961E-05
		^{103}Rh	2.6478E-05
		^{131}Xe	1.8747E-05
		^{133}Cs	5.0289E-05
		^{143}Nd	3.2889E-05
		^{145}Nd	2.7419E-05
		^{147}Pm	6.8639E-06
		^{152}Sm	4.6994E-06
U_3.7:48	Uranium fuel with U ^{235} enrichment 3.7% Wt. corresponding to FA average burnup 48 MW-day/kg HM.	^{235}U	1.3355E-04
		^{236}U	1.0327E-04
		^{238}U	1.9719E-02
		^{237}Np	1.5647E-05
		^{238}Pu	6.7749E-06
		^{239}Pu	1.2662E-04
		^{240}Pu	6.4340E-05
		^{241}Pu	3.4722E-05
		^{242}Pu	1.9037E-05
		^{241}Am	9.8673E-07
		^{16}O	4.2530E-02
		^{135}Xe	6.7806E-09
		^{149}Sm	7.5982E-08
		^{151}Sm	5.2231E-07
		^{99}Tc	5.4515E-05
		^{103}Rh	3.0037E-05
		^{131}Xe	2.0617E-05
		^{133}Cs	5.7961E-05
		^{143}Nd	3.5612E-05
		^{145}Nd	3.1437E-05
		^{147}Pm	6.8134E-06
		^{152}Sm	5.3452E-06

Table B-1 (continued)

Material Name	Comments	Material isotopic composition (isotope, nuclear concentration [10^{-24} cm^{-3}])	
PU_3.6:17	MOX fuel with fissile plutonium isotopes enrichment 3.62% Wt. corresponding to FA average burnup 17 MW-day/kg HM.	^{235}U	3.0534E-05
		^{236}U	2.5385E-06
		^{238}U	2.0144E-02
		^{237}Np	2.4045E-06
		^{238}Pu	1.3292E-06
		^{239}Pu	4.7406E-04
		^{240}Pu	1.4795E-04
		^{241}Pu	5.7132E-05
		^{242}Pu	9.8236E-06
		^{241}Am	1.3594E-06
		^{16}O	4.2506E-02
		^{135}Xe	1.4404E-08
		^{149}Sm	1.4783E-07
		^{151}Sm	7.7056E-07
		^{99}Tc	2.2348E-05
		^{103}Rh	2.2129E-05
		^{131}Xe	1.2186E-05
		^{133}Cs	2.4904E-05
		^{143}Nd	1.5770E-05
		^{145}Nd	1.1215E-05
		^{147}Pm	5.0355E-06
		^{152}Sm	3.2199E-06
PU_3.6:33	MOX fuel with fissile plutonium isotopes enrichment 3.62% Wt. corresponding to FA average burnup 33 MW-day/kg HM.	^{235}U	2.0186E-05
		^{236}U	4.2696E-06
		^{238}U	1.9894E-02
		^{237}Np	4.2797E-06
		^{238}Pu	2.7311E-06
		^{239}Pu	2.9852E-04
		^{240}Pu	1.7846E-04
		^{241}Pu	8.3282E-05
		^{242}Pu	2.5860E-05
		^{241}Am	2.9942E-06
		^{16}O	4.2506E-02
		^{135}Xe	1.1444E-08
		^{149}Sm	1.2062E-07
		^{151}Sm	7.6447E-07
		^{99}Tc	4.0862E-05
		^{103}Rh	3.6232E-05
		^{131}Xe	1.9694E-05
		^{133}Cs	4.4872E-05
		^{143}Nd	2.7603E-05
		^{145}Nd	2.0679E-05
		^{147}Pm	6.6403E-06
		^{152}Sm	5.2658E-06

Table B-1 (continued)

Material Name	Comments	Material isotopic composition (isotope, nuclear concentration [10^{-24} cm^{-3}])	
PU_3.6:41	MOX fuel with fissile plutonium isotopes enrichment 3.62% Wt. corresponding to FA average burnup 41 MW-day/kg HM.	^{235}U	1.5756E-05
		^{236}U	4.8775E-06
		^{238}U	1.9758E-02
		^{237}Np	5.0614E-06
		^{238}Pu	3.8941E-06
		^{239}Pu	2.4228E-04
		^{240}Pu	1.7770E-04
		^{241}Pu	8.7144E-05
		^{242}Pu	3.6474E-05
		^{241}Am	3.4649E-06
		^{16}O	4.2506E-02
		^{135}Xe	1.0225E-08
		^{149}Sm	1.0909E-07
		^{151}Sm	7.5952E-07
		^{99}Tc	4.9207E-05
		^{103}Rh	4.1094E-05
		^{131}Xe	2.2177E-05
		^{133}Cs	5.3602E-05
		^{143}Nd	3.2226E-05
		^{145}Nd	2.5003E-05
		^{147}Pm	6.8821E-06
		^{152}Sm	5.9306E-06
PU_2.7:17	MOX fuel with fissile plutonium isotopes enrichment 2.69% Wt. corresponding to FA average burnup 17 MW-day/kg HM.	^{235}U	2.8612E-05
		^{236}U	2.7944E-06
		^{238}U	2.0347E-02
		^{237}Np	2.4008E-06
		^{238}Pu	1.0993E-06
		^{239}Pu	3.3450E-04
		^{240}Pu	1.2215E-04
		^{241}Pu	4.9442E-05
		^{242}Pu	9.5258E-06
		^{241}Am	1.1240E-06
		^{16}O	4.2508E-02
		^{135}Xe	1.1193E-08
		^{149}Sm	1.1354E-07
		^{151}Sm	5.8719E-07
		^{99}Tc	2.0699E-05
		^{103}Rh	2.0188E-05
		^{131}Xe	1.1163E-05
		^{133}Cs	2.3035E-05
		^{143}Nd	1.4448E-05
		^{145}Nd	1.0429E-05
		^{147}Pm	4.6128E-06
		^{152}Sm	3.0318E-06

Table B-1 (continued)

Material Name	Comments	Material isotopic composition (isotope, nuclear concentration [10^{-24} cm^{-3}])	
PU_2.7:33	MOX fuel with fissile plutonium isotopes enrichment 2.69% Wt. corresponding to FA average burnup 33 MW-day/kg HM.	^{235}U	1.7606E-05
		^{236}U	4.5372E-06
		^{238}U	2.0087E-02
		^{237}Np	4.2361E-06
		^{238}Pu	2.5148E-06
		^{239}Pu	2.1853E-04
		^{240}Pu	1.4136E-04
		^{241}Pu	6.9024E-05
		^{242}Pu	2.5765E-05
		^{241}Am	2.3584E-06
		^{16}O	4.2508E-02
		^{135}Xe	9.0351E-09
		^{149}Sm	9.8177E-08
		^{151}Sm	6.0027E-07
		^{99}Tc	3.7403E-05
		^{103}Rh	3.2327E-05
		^{131}Xe	1.7704E-05
		^{133}Cs	4.0983E-05
		^{143}Nd	2.4621E-05
		^{145}Nd	1.9007E-05
		^{147}Pm	5.9544E-06
		^{152}Sm	4.7872E-06
PU_2.7:41	MOX fuel with fissile plutonium isotopes enrichment 2.69% Wt. corresponding to FA average burnup 41 MW-day/kg HM.	^{235}U	1.3235E-05
		^{236}U	5.0859E-06
		^{238}U	1.9946E-02
		^{237}Np	4.9874E-06
		^{238}Pu	3.6169E-06
		^{239}Pu	1.8670E-04
		^{240}Pu	1.3876E-04
		^{241}Pu	7.0776E-05
		^{242}Pu	3.5686E-05
		^{241}Am	2.6500E-06
		^{16}O	4.2508E-02
		^{135}Xe	8.2684E-09
		^{149}Sm	9.1543E-08
		^{151}Sm	6.1237E-07
		^{99}Tc	4.4969E-05
		^{103}Rh	3.6456E-05
		^{131}Xe	1.9870E-05
		^{133}Cs	4.8863E-05
		^{143}Nd	2.8492E-05
		^{145}Nd	2.2936E-05
		^{147}Pm	6.1528E-06
		^{152}Sm	5.3697E-06

Table B-1 (continued)

Material Name	Comments	Material isotopic composition (isotope, nuclear concentration [10^{-24} cm^{-3}])	
PU_2.4:17	MOX fuel with fissile plutonium isotopes enrichment 2.42% Wt. corresponding to FA average burnup 17 MW-day/kg HM.	^{235}U	2.7777E-05
		^{236}U	2.9076E-06
		^{238}U	2.0405E-02
		^{237}Np	2.3897E-06
		^{238}Pu	1.0335E-06
		^{239}Pu	2.9332E-04
		^{240}Pu	1.1497E-04
		^{241}Pu	4.6985E-05
		^{242}Pu	9.6174E-06
		^{241}Am	1.0464E-06
		^{16}O	4.2508E-02
		^{135}Xe	1.0181E-08
		^{149}Sm	1.0281E-07
		^{151}Sm	5.3010E-07
		^{99}Tc	2.0326E-05
		^{103}Rh	1.9698E-05
		^{131}Xe	1.0916E-05
		^{133}Cs	2.2608E-05
		^{143}Nd	1.4108E-05
		^{145}Nd	1.0254E-05
		^{147}Pm	4.5137E-06
		^{152}Sm	2.9973E-06
PU_2.4:33	MOX fuel with fissile plutonium isotopes enrichment 2.42% Wt. corresponding to FA average burnup 33 MW-day/kg HM.	^{235}U	1.6391E-05
		^{236}U	4.6650E-06
		^{238}U	2.0140E-02
		^{237}Np	4.1976E-06
		^{238}Pu	2.4576E-06
		^{239}Pu	1.9329E-04
		^{240}Pu	1.3072E-04
		^{241}Pu	6.4083E-05
		^{242}Pu	2.6319E-05
		^{241}Am	2.1281E-06
		^{16}O	4.2508E-02
		^{135}Xe	8.2152E-09
		^{149}Sm	9.0532E-08
		^{151}Sm	5.4672E-07
		^{99}Tc	3.6699E-05
		^{103}Rh	3.1334E-05
		^{131}Xe	1.7234E-05
		^{133}Cs	4.0174E-05
		^{143}Nd	2.3798E-05
		^{145}Nd	1.8670E-05
		^{147}Pm	5.7880E-06
		^{152}Sm	4.6882E-06

Table B-1 (continued)

Material Name	Comments	Material isotopic composition (isotope, nuclear concentration [10^{-24} cm^{-3}])	
PU_2.4:41	MOX fuel with fissile plutonium isotopes enrichment 2.42% Wt. corresponding to FA average burnup 41 MW-day/kg HM.	^{235}U	1.2052E-05
		^{236}U	5.1841E-06
		^{238}U	1.9997E-02
		^{237}Np	4.9299E-06
		^{238}Pu	3.5329E-06
		^{239}Pu	1.6835E-04
		^{240}Pu	1.2743E-04
		^{241}Pu	6.5059E-05
		^{242}Pu	3.6143E-05
		^{241}Am	2.3528E-06
		^{16}O	4.2508E-02
		^{135}Xe	7.5758E-09
		^{149}Sm	8.5278E-08
		^{151}Sm	5.6313E-07
		^{99}Tc	4.4092E-05
		^{103}Rh	3.5223E-05
		^{131}Xe	1.9306E-05
		^{133}Cs	4.7856E-05
		^{143}Nd	2.7399E-05
		^{145}Nd	2.2509E-05
		^{147}Pm	5.9661E-06
		^{152}Sm	5.2485E-06
TVEG_4 TVEG_4:0	Uranium-gadolinium fuel with the enrichment 3.6% wt. on ^{235}U and 4 % wt. on Gd_2O_3	^{235}U	7.3225E-04
		^{238}U	1.9360E-02
		^{16}O	4.2056E-02
		^{152}Gd	2.5815E-06
		^{154}Gd	2.7772E-05
		^{155}Gd	1.8730E-04
		^{156}Gd	2.5743E-04
		^{157}Gd	1.9553E-04
		^{158}Gd	3.0843E-04
		^{160}Gd	2.6804E-04
TVEG_5 TVEG_5:0	Uranium-gadolinium fuel with the enrichment 3.3% wt. on ^{235}U and 5 % wt. on Gd_2O_3	^{235}U	6.6163E-04
		^{238}U	1.9143E-02
		^{16}O	4.1938E-02
		^{152}Gd	3.2142E-06
		^{154}Gd	3.4579E-05
		^{155}Gd	2.3321E-04
		^{156}Gd	3.2053E-04
		^{157}Gd	2.4346E-04
		^{158}Gd	3.8403E-04
		^{160}Gd	3.3373E-04

Table B-1 (continued)

Material Name	Comments	Material isotopic composition (isotope, nuclear concentration [10^{-24} cm^{-3}])	
TVEG_4:17	Uranium-gadolinium fuel with the enrichment 3.6% wt. on ^{235}U and 4 % wt. on Gd_2O_3 corresponding to average FA burnup 17 MW-day/kg HM	^{235}U	5.4783E-04
		^{236}U	3.9989E-05
		^{238}U	1.9139E-02
		^{237}Np	3.7973E-06
		^{238}Pu	4.9031E-07
		^{239}Pu	1.2109E-04
		^{240}Pu	1.7377E-05
		^{241}Pu	7.0208E-06
		^{242}Pu	5.4476E-07
		^{241}Am	8.8029E-08
		^{16}O	4.2055E-02
		^{152}Gd	1.8321E-06
		^{154}Gd	2.5080E-05
		^{155}Gd	1.0215E-06
		^{156}Gd	4.3334E-04
		^{157}Gd	2.5976E-07
		^{158}Gd	5.0718E-04
		^{160}Gd	2.6656E-04
		^{135}Xe	1.0322E-08
		^{149}Sm	1.0735E-07
		^{151}Sm	4.0519E-07
		^{99}Tc	1.2602E-05
		^{103}Rh	7.9971E-06
		^{131}Xe	5.9379E-06
		^{133}Cs	1.3779E-05
		^{143}Nd	1.0973E-05
		^{145}Nd	7.6416E-06
		^{147}Pm	3.2794E-06
		^{152}Sm	1.2631E-06

Table B-1 (continued)

Material Name	Comments	Material isotopic composition (isotope, nuclear concentration [10^{-24} cm^{-3}])	
TVEG_4:33	Uranium-gadolinium fuel with the enrichment 3.6% wt. on ^{235}U and 4 % wt. on Gd_2O_3 corresponding to average FA burnup 33 MW-day/kg HM	^{235}U	3.4998E-04
		^{236}U	7.3537E-05
		^{238}U	1.8901E-02
		^{237}Np	9.6340E-06
		^{238}Pu	2.5298E-06
		^{239}Pu	1.5184E-04
		^{240}Pu	4.4993E-05
		^{241}Pu	2.4072E-05
		^{242}Pu	5.0595E-06
		^{241}Am	5.5623E-07
		^{16}O	4.2055E-02
		^{152}Gd	1.0821E-06
		^{154}Gd	2.2136E-05
		^{155}Gd	1.6393E-07
		^{156}Gd	4.2171E-04
		^{157}Gd	2.0690E-07
		^{158}Gd	5.1162E-04
		^{160}Gd	2.6501E-04
		^{135}Xe	9.6832E-09
		^{149}Sm	1.0224E-07
		^{151}Sm	5.0972E-07
		^{99}Tc	2.9713E-05
		^{103}Rh	1.8557E-05
		^{131}Xe	1.2977E-05
		^{133}Cs	3.2177E-05
		^{143}Nd	2.3818E-05
		^{145}Nd	1.7521E-05
		^{147}Pm	5.5753E-06
		^{152}Sm	3.0576E-06

Table B-1 (continued)

Material Name	Comments	Material isotopic composition (isotope, nuclear concentration [10^{-24} cm^{-3}])	
TVEG_4:41	Uranium-gadolinium fuel with the enrichment 3.6% wt. on ^{235}U and 4 % wt. on Gd_2O_3 corresponding to average FA burnup 41 MW-day/kg HM	^{235}U	2.6926E-4
		^{236}U	8.5042E-5
		^{238}U	1.8773E-2
		^{237}Np	1.2814E-5
		^{238}Pu	4.3110E-6
		^{239}Pu	1.5300E-4
		^{240}Pu	5.7222E-5
		^{241}Pu	3.1551E-5
		^{242}Pu	9.3635E-6
		^{241}Am	8.5318E-7
		^{16}O	4.2055E-2
		^{152}Gd	7.9513E-7
		^{154}Gd	2.0634E-5
		^{155}Gd	1.4382E-7
		^{156}Gd	4.1502E-4
		^{157}Gd	1.8693E-7
		^{158}Gd	5.1379E-4
		^{160}Gd	2.6417E-4
		^{135}Xe	9.0996E-9
		^{149}Sm	9.6559E-8
		^{151}Sm	5.5174E-7
		^{99}Tc	3.8089E-5
		^{103}Rh	2.3356E-5
		^{131}Xe	1.5872E-5
		^{133}Cs	4.1001E-5
		^{143}Nd	2.9142E-5
		^{145}Nd	2.2213E-5
		^{147}Pm	6.1071E-6
		^{152}Sm	3.8311E-6

Table B-1 (continued)

Material Name	Comments	Material isotopic composition (isotope, nuclear concentration [10^{-24} cm^{-3}])	
TVEG_5:15	Uranium-gadolinium fuel with the enrichment 3.3% wt. on ^{235}U and 5 % wt. on Gd_2O_3 corresponding to average FA burnup 17 MW-day/kg HM	^{235}U	4.8382E-04
		^{236}U	3.5164E-05
		^{238}U	1.8968E-02
		^{237}Np	2.5863E-06
		^{238}Pu	2.9960E-07
		^{239}Pu	8.8781E-05
		^{240}Pu	1.5353E-05
		^{241}Pu	6.2524E-06
		^{242}Pu	6.0179E-07
		^{241}Am	6.9561E-08
		^{16}O	4.1938E-02
		^{152}Gd	2.2242E-06
		^{154}Gd	3.1600E-05
		^{155}Gd	3.2385E-07
		^{156}Gd	5.4422E-04
		^{157}Gd	1.8330E-07
		^{158}Gd	6.3019E-04
		^{160}Gd	3.3229E-04
		^{135}Xe	8.4538E-09
		^{149}Sm	8.0821E-08
		^{151}Sm	3.1290E-07
		^{99}Tc	1.2031E-05
		^{103}Rh	7.4977E-06
		^{131}Xe	5.7195E-06
		^{133}Cs	1.3175E-05
		^{143}Nd	1.0380E-05
		^{145}Nd	7.3069E-06
		^{147}Pm	3.3064E-06
		^{152}Sm	1.2690E-06

Table B-1 (continued)

Material Name	Comments	Material isotopic composition (isotope, nuclear concentration [10^{-24} cm^{-3}])	
TVEG_5:32	Uranium-gadolinium fuel with the enrichment 3.3% wt. on ^{235}U and 5 % wt. on Gd_2O_3 corresponding to average FA burnup 32 MW-day/kg HM	^{235}U	2.6776E-04
		^{236}U	7.0094E-05
		^{238}U	1.8728E-02
		^{237}Np	7.7190E-06
		^{238}Pu	2.0282E-06
		^{239}Pu	1.1559E-04
		^{240}Pu	4.1294E-05
		^{241}Pu	2.2283E-05
		^{242}Pu	6.3481E-06
		^{241}Am	4.9976E-07
		^{16}O	4.1938E-02
		^{152}Gd	1.1130E-06
		^{154}Gd	2.7446E-05
		^{155}Gd	1.5769E-07
		^{156}Gd	5.3058E-04
		^{157}Gd	1.6774E-07
		^{158}Gd	6.3494E-04
		^{160}Gd	3.3037E-04
		^{135}Xe	7.4945E-09
		^{149}Sm	8.0584E-08
		^{151}Sm	4.0538E-07
		^{99}Tc	3.0625E-05
		^{103}Rh	1.8625E-05
		^{131}Xe	1.3349E-05
		^{133}Cs	3.3172E-05
		^{143}Nd	2.3428E-05
		^{145}Nd	1.8017E-05
		^{147}Pm	5.7735E-06
		^{152}Sm	3.2490E-06

Table B-1 (continued)

Material Name	Comments	Material isotopic composition (isotope, nuclear concentration [10^{-24} cm^{-3}])	
TVEG_5:40	Uranium-gadolinium fuel with the enrichment 3.3% wt. on ^{235}U and 5 % wt. on Gd_2O_3 corresponding to average FA burnup 40 MW-day/kg HM	^{235}U	1.9449E-04
		^{236}U	8.0032E-05
		^{238}U	1.8603E-02
		^{237}Np	1.0347E-05
		^{238}Pu	3.5111E-06
		^{239}Pu	1.1801E-04
		^{240}Pu	5.1119E-05
		^{241}Pu	2.8043E-05
		^{242}Pu	1.1103E-05
		^{241}Am	7.2654E-07
		^{16}O	4.1938E-02
		^{152}Gd	7.6549E-07
		^{154}Gd	2.5465E-05
		^{155}Gd	1.4181E-07
		^{156}Gd	5.2333E-04
		^{157}Gd	1.5847E-07
		^{158}Gd	6.3726E-04
		^{160}Gd	3.2938E-04
		^{135}Xe	6.9699E-09
		^{149}Sm	7.7330E-08
		^{151}Sm	4.4104E-07
		^{99}Tc	3.8586E-05
		^{103}Rh	2.2940E-05
		^{131}Xe	1.5989E-05
		^{133}Cs	4.1526E-05
		^{143}Nd	2.7732E-05
		^{145}Nd	2.2419E-05
		^{147}Pm	6.1251E-06
		^{152}Sm	3.9837E-06

Table B-1 (continued)

Material Name	Comments	Material isotopic composition (isotope, nuclear concentration [10^{-24} cm^{-3}])	
TVEG_5:48	Uranium-gadolinium fuel with the enrichment 3.3% wt. on ^{235}U and 5 % wt. on Gd_2O_3 corresponding to average FA burnup 48 MW-day/kg HM	^{235}U	1.3690E-04
		^{236}U	8.6439E-05
		^{238}U	1.8470E-02
		^{237}Np	1.2843E-05
		^{238}Pu	5.3772E-06
		^{239}Pu	1.1762E-04
		^{240}Pu	5.9015E-05
		^{241}Pu	3.2177E-05
		^{242}Pu	1.6795E-05
		^{241}Am	9.0181E-07
		^{16}O	4.1938E-02
		^{152}Gd	5.0751E-07
		^{154}Gd	2.3480E-05
		^{155}Gd	1.2682E-07
		^{156}Gd	5.1558E-04
		^{157}Gd	1.4943E-07
		^{158}Gd	6.3960E-04
		^{160}Gd	3.2832E-04
		^{135}Xe	6.4707E-09
		^{149}Sm	7.3544E-08
		^{151}Sm	4.7089E-07
		^{99}Tc	4.5989E-05
		^{103}Rh	2.6571E-05
		^{131}Xe	1.8036E-05
		^{133}Cs	4.9140E-05
		^{143}Nd	3.0894E-05
		^{145}Nd	2.6414E-05
		^{147}Pm	6.2013E-06
		^{152}Sm	4.6204E-06

Table B-1 (continued)

Material Name	Comments	Material isotopic composition (isotope, nuclear concentration [10^{-24} cm^{-3}])	
General material type: Non-fuel (Subtype: Clad)			
CL1	Cladding of a fuel pin (tveg)	Zr Nb Hf	4.257E-02 4.223E-04 6.594E-06
CL2	Cladding of an absorbing pin	Fe Cr Ni Ti C	5.933E-02 1.687E-02 8.477E-03 9.904E-04 4.737E-04
General material type: Non fuel (Subtype: Absorber)			
B_4C_nat	Material of an absorbing pin from boron carbide (B_4C) on the base of natural boron	B^{10} B^{11} C	1.555E-02 6.300E-02 1.964E-02
B_4C_enr	Material of an absorbing pin from boron carbide (B_4C) on the base of enriched boron (80% of ^{10}B)	B^{10} B^{11} C	6.571E-02 1.643E-02 2.053E-02
BA_rod	Material of a burnable poison pin with the natural boron content 0.065 g/cm^3	B^{10} B^{11} Al Fe Ni Cr Zr	6.897E-04 2.798E-03 5.613E-02 5.663E-05 5.382E-04 1.747E-03 3.465E-04

Table B-1 (continued)

Material Name	Comments	Material isotopic composition (isotope, nuclear concentration [10^{-24} cm^{-3}])	
General material type:Non fuel (Subtype: Moderator)			
M575B1.3	Moderator with the boron content 1300 ppm, T_m=575K, γ= 0.7241 g/cm^3	H ^{16}O B^{10} B^{11}	4.8410E-02 2.4205E-02 1.0381E-05 4.2049E-05
M575B0.6	Moderator with the boron content 600 ppm, T_m=575K, γ= 0.7241 g/cm^3	H ^{16}O B^{10} B^{11}	4.8410E-02 2.4205E-02 4.7913E-06 1.9407E-05
M575B0	Moderator without boron, T_m=575K, γ= 0.7241 g/cm^3	H ^{16}O B^{10} B^{11}	4.8410E-02 2.4205E-02 0.0 0.0
M560B1.3	Moderator with the boron content 1300 ppm, T_m=560K, γ= 0.7533 g/cm^3	H ^{16}O B^{10} B^{11}	5.0362E-02 2.5181E-02 1.0800E-05 4.3744E-05
M560B0.6	Moderator with the boron content 600 ppm, T_m=560K, γ= 0.7533 g/cm^3	H ^{16}O B^{10} B^{11}	5.0362E-02 2.5181E-02 4.9845E-06 2.0190E-05
M560B0	Moderator without boron, T_m=560K, γ= 0.7533 g/cm^3	H ^{16}O B^{10} B^{11}	5.0362E-02 2.5181E-02 0.0 0.0
M553B0	Moderator without boron, T_m=553K, γ= 0.7657 g/cm^3	H ^{16}O B^{10} B^{11}	5.1192E-02 2.5596E-02 0.0 0.0
M300B2.8	Moderator with the boron content 2800 ppm, T_m=300K, γ= 1.0033 g/cm^3	H ^{16}O B^{10} B^{11}	6.7076E-02 3.3538E-02 3.0981E-05 1.2549E-04
M300B0.6	Moderator with the boron content 600 ppm, T_m=300K, γ= 1.0033 g/cm^3	H ^{16}O B^{10} B^{11}	6.7076E-02 3.3538E-02 6.6387E-06 2.6890E-05
M300B0	Moderator without boron, T_m=300K, γ= 1.0033 g/cm^3	H ^{16}O B^{10} B^{11}	6.7076E-02 3.3538E-02 0.0 0.0
M575/0.2/B0	Moderator without boron, T_m=575, γ= 0.2 g/cm^3	H ^{16}O B^{10} B^{11}	1.3371E-02 6.6855E-03 0.0 0.0

Table B-2 Isotopic composition of certain materials

Isotopic content of certain materials (nateral mixture), atomic percent

Zr	Zr- 90	Zr- 91	Zr- 92	Zr- 94	Zr- 96			Σ
	51.5	11.2	17.1	17.4	2.8			100.000
Nb	Nb- 93							Σ
	100							100.000
Hf	Hf- 174	Hf- 176	Hf- 177	Hf -178	Hf -179	Hf- 180		Σ
	0.162	5.206	18.606	27.297	13.629	35.1		100.000
Fe	Fe –54	Fe -56	Fe -57	Fe -58				Σ
	5.9	91.72	2.1	0.28				100.000
Cr	Cr –50	Cr -52	Cr -53	Cr -54				Σ
	4.345	83.789	9.501	2.365				100.000
Ni	Ni –58	Ni -60	Ni -61	Ni -62	Ni -64			Σ
	68.077	26.223	1.14	3.634	0.926			100.000
Ti	Ti –46	Ti -47	Ti -48	Ti -49	Ti -50			Σ
	8	7.3	73.8	5.5	5.4			100.000
C	C –12	C -13						Σ
	98.9	1.1						100.000
Al	Al –27							Σ
	100							100.000
Gd	Gd-152	Gd-154	Gd-155	Gd-156	Gd-157	Gd-158	Gd-160	Σ
	0.196	2.155	14.73	20.47	15.675	24.873	21.901	100.000
Plutonium isotopic composition of MOX fuel, % wt.								
Plutonium	Pu-238	Pu-239	Pu-240	Pu-241	Pu-242			Σ
	0.13	91.72	6.55	1.17	0.43			100.000
Isotopic composition of Zirconium alloy, % wt.								
Zirconium alloy	Zr	Nb	Hf					Σ
	98.97	1.0	0.03					100.000

Remark:

Weight enrichment by fissile nuclides is related to a total weight of a heavy metal defining a fuel. Weight enrichment by Gd_2O_3 is related to a total weight of a fuel (including an oxygen and Gd_2O_3).

In nuclear concentration calculation Avogadro number is equal to $0.60221*10^{23}$.

Geometry description

Let all the local coordinates systems, in which different types of geometrical objects are described, be positioned in the similar manner. Include the following definitions:

- *Zone (global description type of a homogeneous spatial region)* – Elements of this set are to be used for a description of similar geometrical objects that are characterized by a certain type of covering surface (in local coordinates), a certain material and a certain temperature. The type *Zone* can include a set of subtypes (subsets of geometrical objects). For example the zones possessing the same outer cylindrical surface (Zone: Cylinder:) or the zones of cylindrical layer type could be defined as the subtypes of the general "zone:" type.
- *Cell (global description type of a heterogeneous spatial region consisting of a regular set of Zone type elements)* – Elements of this type set are to be used for a description of similar spatial regions characterized by a definite type of covering surface (in local coordinates) and by a regular – over their mutual surfaces - set of elements of Zone or Cell types. It should be noted that a regulation of inner zones of Cell type objects means a regulation of zones local coordinates towards a local coordinates system of Cell type object;
- *Collection* (abstract data type) - it is a set combining different types and subtypes in a new abstract type.
- *Collection: Moderator_zones* – it is a set of zones containing a material of *Moderator* type.
- *Collection:**Absorber_zones*** – it is a set of zones containing a material of *Absorber* type.
- *Collection:Non_fuel_zones* – it is a set of zones of all types containing a non-fuel material (as *Non_fuel: material)*. This set contains *Collection: Moderator_zones*, *Collection:**Absorber**_zones* etc.

- *Cell:**HEX1:*** – Cell with a hexagonal outer surface containing a set of different zones or cells with a cylindrical outer surface. The orientation of the outer surface of this type objects is presented in Fig. B-5;

- *Cell:**HEX2:*** - A regulated set of zones or cells with a hexagonal outer surface. The orientation of the outer surface of this type objects is presented in Fig. B-5 (see the shaded area);

- *Cell:**HEX1_SET*** – Data type describing a spatial region of regulated objects of the type ***HEX1:*** (see Fig. B-5);

- *Cell:**Assembly*** –Structure characterizing a spatial region with a hexagonal outer border (HEX2) that is designed by a crossing of a HEX2 type cell with a ***HEX1_SET*** type object (Fig. B-5). It should be noted that from one side *Assembly* type object differs from ***HEX1_SET*** type object by the presence of gap cells with the outer border defined by the crossing of HEX1 and HEX2 type objects and from the other side the outer border is completely defined by ***HEX1_SET*** type object because from geometrical point of view *Assembly* type object is the subtype of ***HEX1_SET*** type object. So while defining sub-objects of *Assembly* type we will define a ***HEX1_SET*** type object;
- *Cell:Minicore* as a geometrical object type — Structure characterizing a spatial region with a definite type of outer surface. The region includes the regulated - over outer borders – set of the elements of the following types: *Zone, Cell and Assembly* with the coordinates center in *Minicore(**r**)*;

Let Cyl($\mathbf{r}$,$\mathbf{r}_1$) - cylinder with a radius $\mathbf{r}_1$ with the center in $\mathbf{r}$;

Hex($\mathbf{r}$,h=**1.275 cm**) - hexahedron with the center in $\mathbf{r}$ *and the width across flats h=**1.275** cm*;

1. ***HEX: Central cell***

r Cell center

***h*=1,275cm** Hexahedron outer surface

Include

*Moderator_zones{...,**Zone:Moderator in Central cell** ,...}*

[Cyl(***r***, r_l=**0.55 cm**)\Cyl(***r***,r_l=**0. cm**)]U[Hex(***r***,h=**1.275 cm**)\ Cyl(***r***,r_l=**0.63 cm**)]

Mat *Non-fuel:* ***Moderator:***material

Tem Zone temperature

*Non-fuel zones{..., Zone:**Central_tube** ,...}*

[Cyl(***r***, r_l=**0.63 cm**)\Cyl(***r***,r_l=**0.55 cm**)]

Mat ***Non-fuel:***CL1

Tem Zone temperature

*End description: **Central cell***

*2. HEX:**Fuel cell***

r Cell center

***h*=1,275cm** Hexahedron outer surface

Include

*Zone:**Fuel***

[Cyl(***r***,r_l=**0.386 cm**)\Cyl(***r***,r_l=**0. cm**)]

Mat ***Fuel:***material

Tem Zone temperature

*Non-fuel zones{...,Zone:**Clad_fuel** ,...}*

[Cyl(***r***, r_l=**0.455 cm**)\Cyl(***r***,r_l=**0.386 cm**)]

Mat ***Non-fuel:*** CL1

Tem Zone temperature

*Moderator_zones{...,Zone: **Moderator in Fuel cell ,...}***

[Hex(***r***,h=**1.275 cm**)\Cyl(***r***,r_l=**0.455 cm**)]

Mat *Non-fuel*: ***Moderator:***material

Tem Zone temperature

*End description: **Fuel cell***

3. HEX: ***Guide tube cell***

r	Cell center
***h*=1,275cm**	Hexahedron outer surface
Include	

*Cell:**Into Guide tube***	abstract type
r	Cell center
r_l=0.55 cm	Cylinder outer surface

Non-fuel zones{...,Zone: ***Guide tube ,...}***

Cyl(***r***, **r_l=0.63 cm**)\Cyl(***r***,**r_l=0.55 cm**)

Mat	***Non-fuel:***CL1
Tem	Zone temperature

Moderator_zones{ ...,Zone: ***Moderator in Guide tube cell ,...}***

Hex(***r***,h=**1.275 cm**)\Cyl(***r***,**r_l=0.63 cm**)

Mat	*Non-fuel:* ***Moderator:*** material
Tem	Zone temperature

End description: ***Guide tube cell***

3.1 ***Into Guide tube:Mod_cell***

r	Cell center
r_l=0.55 cm	Cylinder outer surface
Include	

*Moderator_zones{...,Zone:****Moderator in Mod_cell ,...}***

[Cyl(***r***, **r_l=0.55 cm**)\Cyl(***r***,**r_l=0. cm**)]

Mat	*Non-fuel:**Moderator:***material
Tem	Zone temperature

End description: ***Moderator_cell***

3.2 ***Into Guide tube:CRcell***

r Cell center

r_1**=0.55 cm** Cylinder outer surface

Include

Absorber zones {..,Zone: ***CR rod ,...}***

[Cyl(r,r_1=**0.35 cm**)\Cyl(r,r_1=**0. cm**)]

Mat *Non-fuel:****Absorber:*** material

Tem Zone temperature

*Non-fuel zones{...,Zone:****CR rod Clad ,...}***

[Cyl(r, r_1=**0.41 cm**)\Cyl(r,r_1=**0.35 cm**)]

Mat ***Non-fuel:***CL2

Tem Zone temperature

*Moderator_zones{...,Zone:****Moderator in CRcell ,...}***

[Cyl(r, r_1=**0.55 cm**)\Cyl(r,r_1=**0.41 cm**)]

Mat *Non-fuel:* ***Moderator:***material

Tem Zone temperature

End description: ***CRcell***

3.3 ***Into Guide tube :BA cell***

r Cell center

r_1**=0.55 cm** Cylinder outer surface

Include

Absorber zones {..,Zone: ***BA rod ,...}***

[Cyl(r,r_1=**0.386 cm**)\Cyl(r,r_1=**0. cm**)]

Mat *Non-fuel:****Absorber:***material

Tem Zone temperature

Non-fuel zones{...,Zone:BA ***rod Clad ,...}***

[Cyl(r, r_1=**0.455 cm**)\Cyl(r,r_1=**0.386 cm**)]

Mat ***Non-fuel:*** CL1

Tem Zone temperature

*Moderator_zones{ ...,Zone:****Moderator in BAcell ,...}***

[Cyl(r, r_1=**0.55 cm**)\Cyl(r,r_1=**0.455 cm**)]

Mat *Non-fuel:* ***Moderator:***material

Tem Zone temperature

End description: ***BA cell***

*4. HEX: **Mod_hex1***

r	Cell center
***h*=1,275cm**	Hexahedron outer surface

Include

*Moderator_zones{ ...,Zone:**Mod_hex1_zone**,...}*

***Hex1(h*=1,275cm)**\ Cyl(*r*, r_1=0 cm)

Mat	*Non-fuel: **Moderator:***material
Tem	Zone temperature

*End description: **Mod_hex1***

*5. HEX:Steel_**hex1***

r	Cell center
***h*=1,275cm**	Hexahedron outer surface

Include

*Non-fuel zones { ...,Zone:**Steel_hex1_zone**,...}*

***Hex1(h*=1,275cm)** \ Cyl(*r*, r_1=0 cm)

Mat	*Non-fuel:CL2*
Tem	Zone temperature

*End description: Steel_**hex1***

Table B-3. Description of Fuel Assemblies Types

Assembly:	K1 (Fig. B-1)	K2 (Fig. B-1)
Hex1:1	***Central cell***	***Central cell***
Hex1:2	***Fuel cell*** with ***Fuel:***material=U_3.7	***Fuel cell*** with ***Fuel:***material=PU_3.6
Hex1:3	***Guide tube cell***	***Guide tube cell***
Hex1:4		
Hex1:5		
Hex1:7	***Mod_hex1***	***Mod_hex1***

Assembly:	K3 (Fig. B-2)	K4 (Fig. B-3)
Hex1:1	***Central cell***	***Central cell***
Hex1:2	***Fuel cell*** with ***Fuel:***material=U_4.2	***Fuel cell*** with ***Fuel:***material=PU_3.6
Hex1:3	***Guide tube cell***	***Guide tube cell***
Hex1:4	***Fuel cell*** with ***Fuel:***material=U_4.2	***Fuel cell*** with ***Fuel:***material=PU_3.6
Hex1:5	***Fuel cell*** with ***Fuel:***material=U_3.7	***Fuel cell*** with ***Fuel:***material=PU_2.7
Hex1:6		***Fuel cell*** with ***Fuel:***material=PU_2.4
Hex1:7	***Mod_hex1***	***Mod_hex1***

Assembly:	K5 (Fig. B-2)	K6 (Fig. B-3)
Hex1:1	***Central cell***	***Central cell***
Hex1:2	***Fuel cell*** with ***Fuel:***material=U_4.2	***Fuel cell*** with ***Fuel:***material=PU_3.6
Hex1:3	***Guide tube cell***	***Guide tube cell***
Hex1:4	***Fuel cell*** with ***Fuel:***material=TVEG_5	***Fuel cell*** with ***Fuel:***material=TVEG_4
Hex1:5	***Fuel cell*** with ***Fuel:***material=U_3.7	***Fuel cell*** with ***Fuel:***material=PU_2.7
Hex1:6		***Fuel cell*** with ***Fuel:***material=PU_2.4
Hex1:7	***Mod_hex1***	***Mod_hex1***

Assembly:	K7 (Fig. B-4)	K8 (Fig. B-4)
Hex1: (1-397)	***Mod_hex1***	*Steel_**hex1***

Table B-4. Description of 7 FAs test groups

System Number	1	2	3	4
MiniCore:	***MC U*** (Fig. B-7)	***MC PU1 U6*** (Fig. B-7)	***MC U4^{p} PU3^{p}*** (Fig. B-7)	***MC PUgd1^{p} U6*** (Fig. B-7)
Assembly:1	***K1***	***K2***	***K3***	***K6***
Assembly:2	***K1***	***K1***	***K4***	***K1***
Assembly:3	***K1***	***K1***	***K3***	***K1***
Assembly:4	***K1***	***K1***	***K4***	***K1***
Assembly:5	***K1***	***K1***	***K3***	***K1***
Assembly:6	***K1***	***K1***	***K4***	***K1***
Assembly:7	***K1***	***K1***	***K3***	***K1***
Assembly:8	***K8***	***K8***	***K8***	***K8***
Assembly:9	***K7***	***K7***	***K7***	***K7***

System Number	5	6	7
MiniCore:	***MC PU*** (Fig. B-7)	***MC PUgd1^{p} PU6*** (Fig. B-7)	***MC Ugd4^{p} Pugd3*** (Fig. B-7)
Assembly:1	***K2***	***K6***	***K5***
Assembly:2	***K2***	***K2***	***K6***
Assembly:3	***K2***	***K2***	***K5***
Assembly:4	***K2***	***K2***	***K6***
Assembly:5	***K2***	***K2***	***K5***
Assembly:6	***K2***	***K2***	***K6***
Assembly:7	***K2***	***K2***	***K5***
Assembly:8	***K8***	***K8***	***K8***
Assembly:9	***K7***	***K7***	***K7***

Table B-5. Description of states

State Number	*State name*	Absorber_ zones: Assembly:1 ***Fig. B-7)***	Absorber_ Zones: Assembly:(2-7) ***(Fig. B-7)***	Type: *Fuel* (temp., K)	Type: *Non_Fuel_zones* (temp., K)	Type: *Moderator_zones*	***Assembly:1: Into Guide tube (Fig. B-7)***	^{135}Xe, ^{149}Sm
1	S1	-	-	1027	575	M575B0.6	***Mod_cell***	0.0
2	S2	-	-	575	575	M575B0.6	***Mod_cell***	0.0
3	S21	-	-	575	575	M575B0	***Mod_cell***	0.0
4	S22	-	-	575	575	M575/0.2/B0	***Mod_cell***	0.0
5	S3	-	-	300	300	M300B0.6	***Mod_cell***	0.0
6	S31	-	-	300	300	M300B0	***Mod_cell***	0.0
7	SA1	B_4C (nat)	-	1027	575	M575B0.6	***CRcell***	0.0
8	SA2	B_4C (enr)	-	1027	575	M575B0.6	***CRcell***	0.0
9	SA3	BA_rod	-	1027	575	M575B0.6	***BAcell***	0.0

All calculations should be performed with zero axial leakage.

Fig. B- 1. Pattern of non-graded assembly

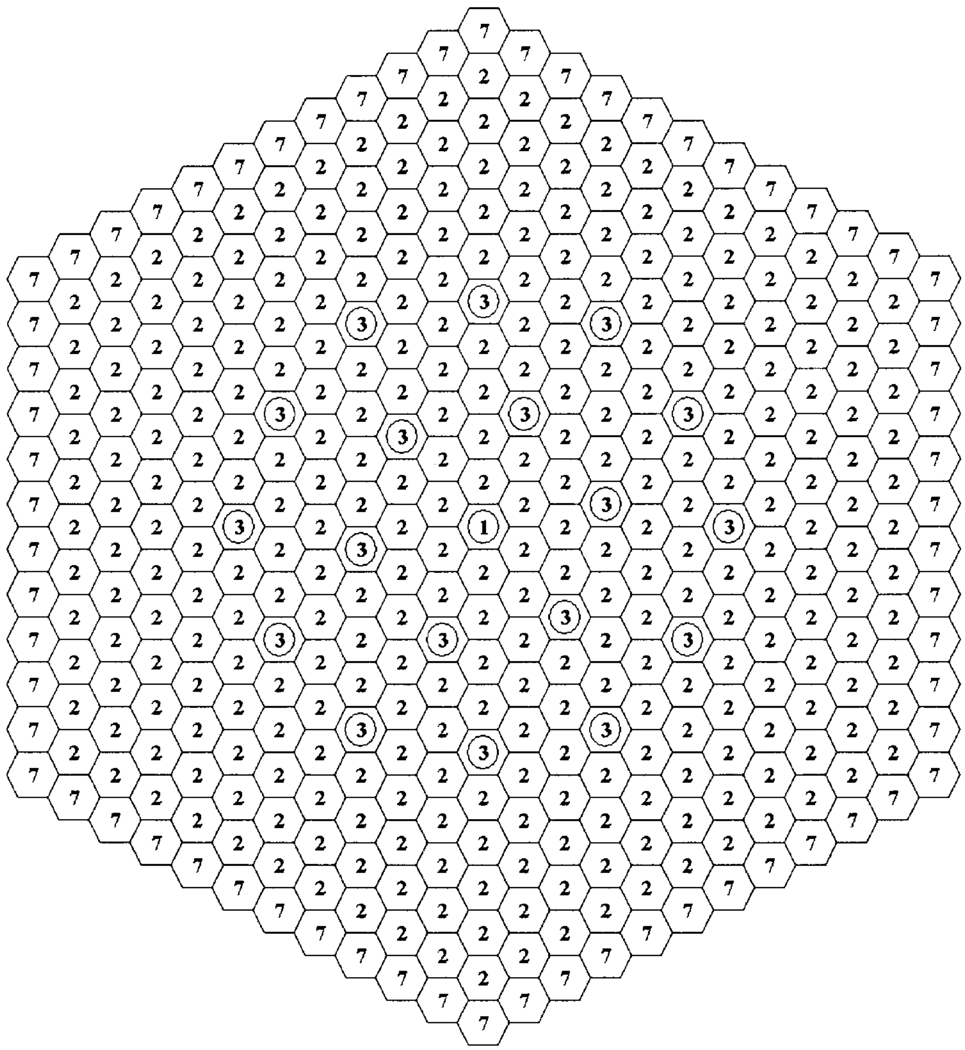


Fig. B- 2. Pattern of graded UOX assembly

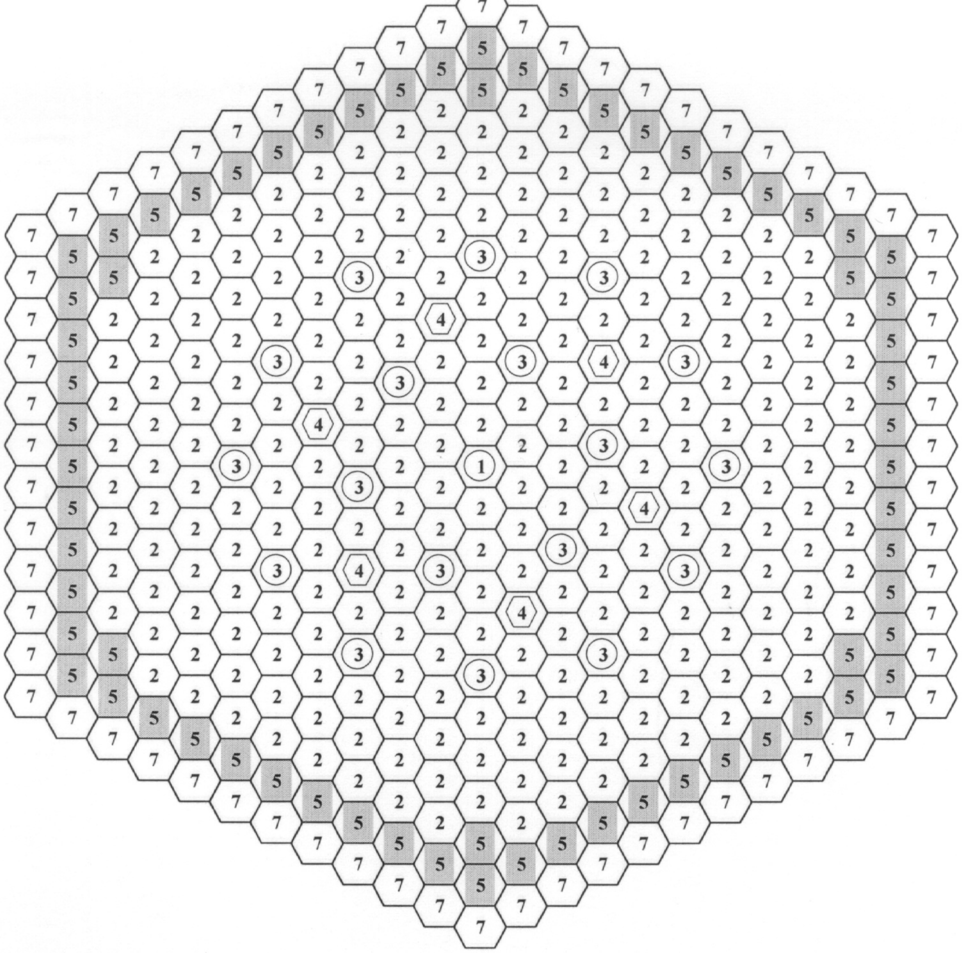


Fig. B- 3. Pattern of graded MOX assembly

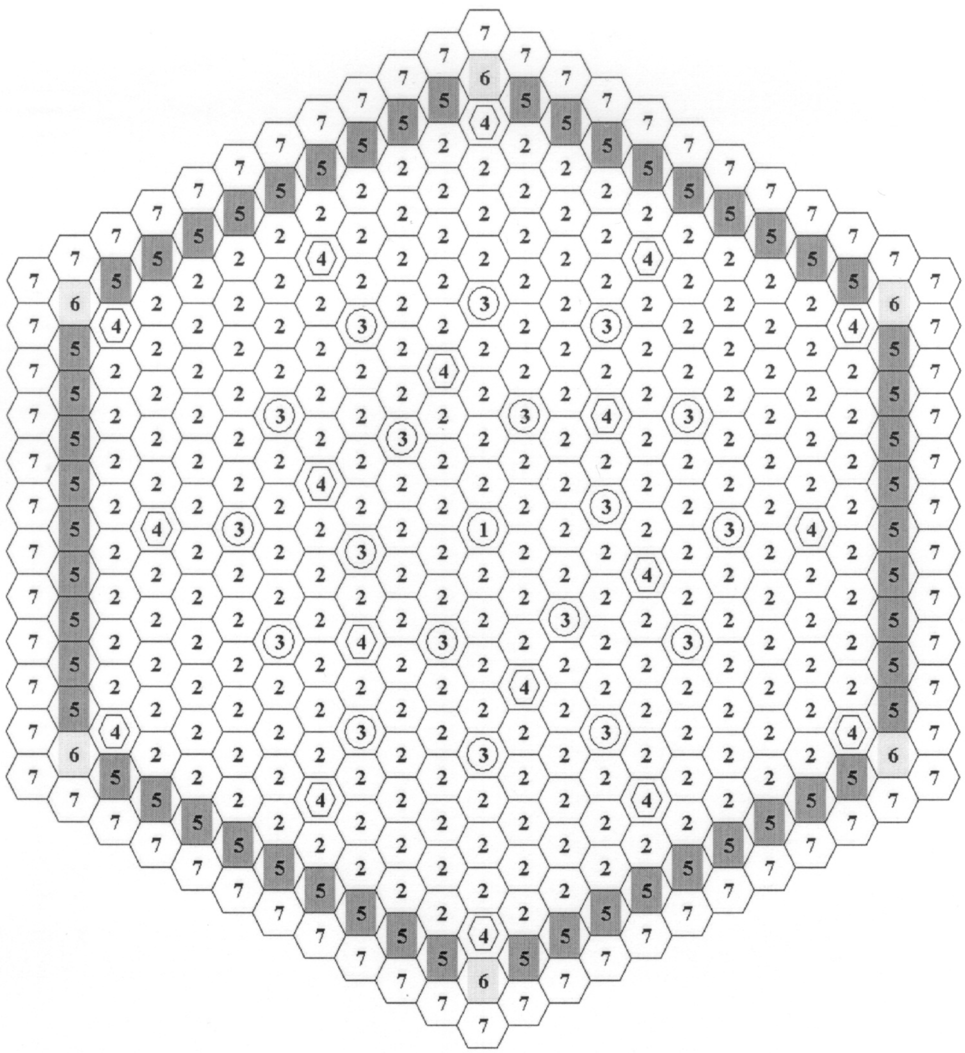


Fig. B-4. Location *HEX*:objects in ***HEX_SETS:object***

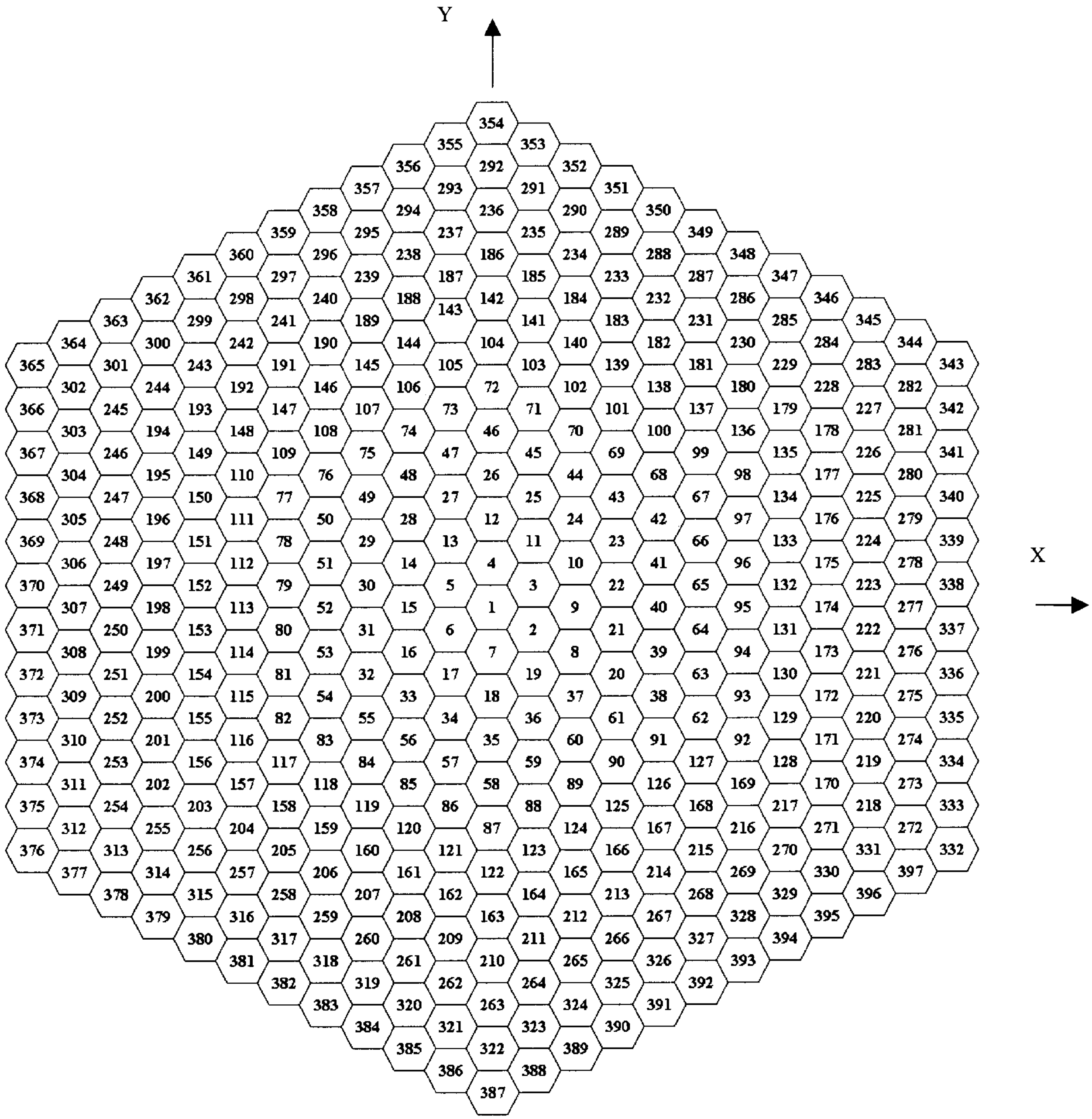


Fig. B-5. Objects location in an ASSEMBLY***:object***

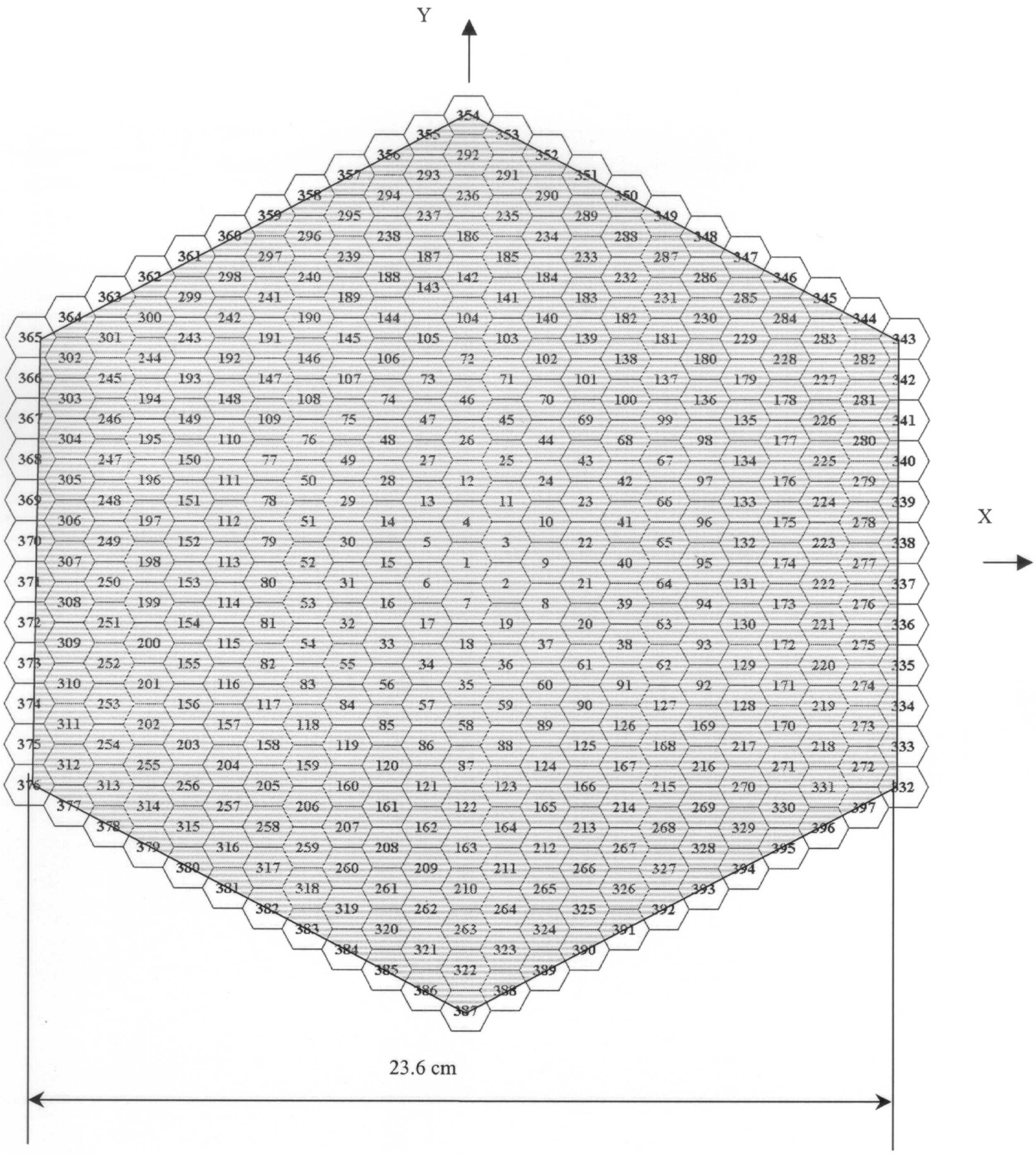


Fig. B-6. Assembly Set

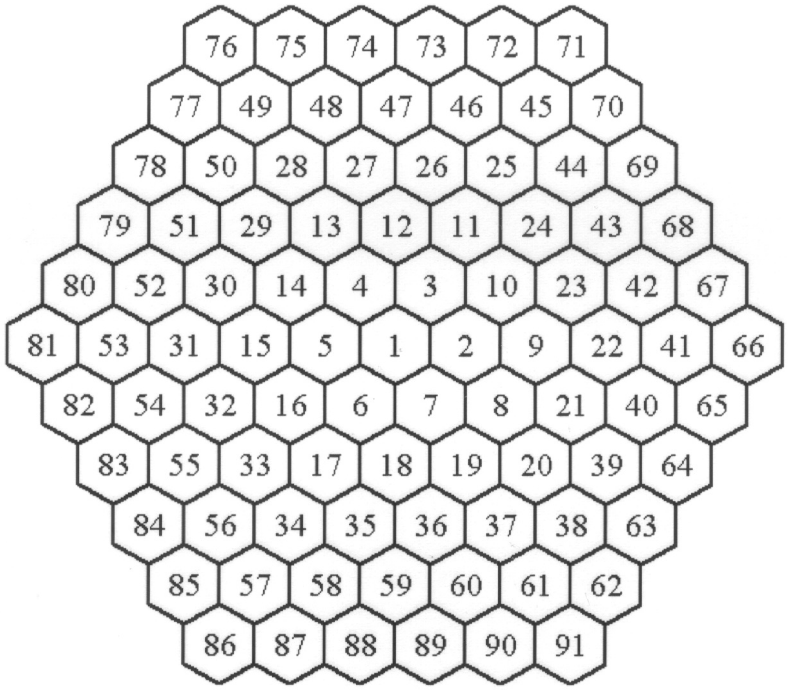


Fig. B-7. Minicore

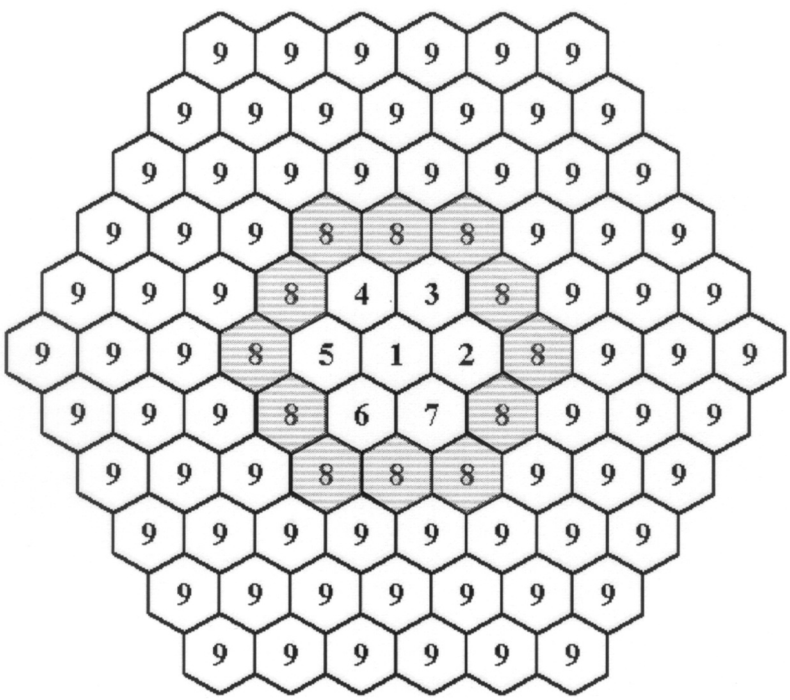
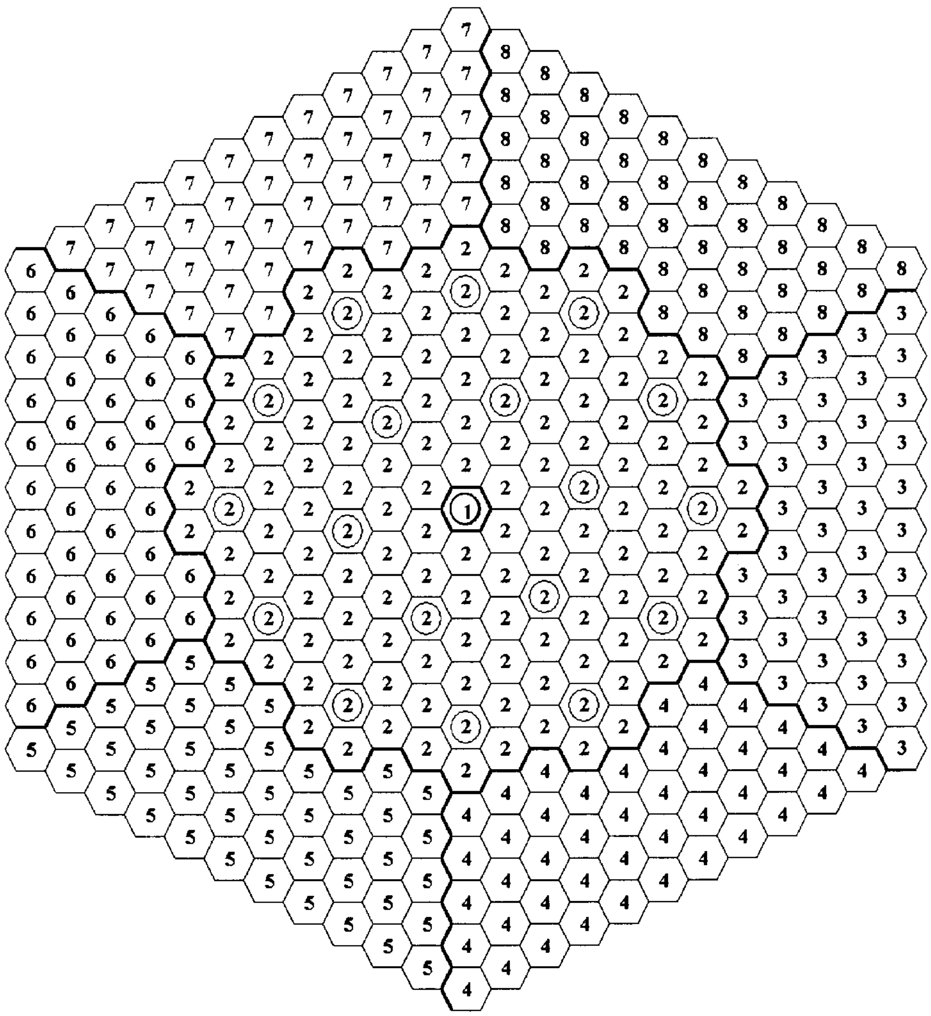


Fig. B-8. Registration areas in FA



Annex C

Fig. C-1, C-2 and C-3 represent the VVER-1000 model in plane. It contains the following objects:

1. Regular grid of fuel assemblies with a water gap (see Fig. C-4) - "Lcom"
   1.1 Regular grid of fuel assemblies (see Fig. C-5) - "L1000"
   1.2 Water gap between the regular FA grid and the steel buffer (gap width of 3mm) (see Fig. C-4) –"W_gap"
2. Group of the holes (cavities) located in the spatial region – [Cyl(r,r=R4)\Lcom] for cooling the steel buffer (geometrical zone -"Zone:HoleV"). The parameters of the holes are presented in the Table in Fig. C-1
3. Steel buffer – [[Cyl(r,r=R4)\Lcom]\HoleV] – "Zone:V".
4. Steel reactor barrel – [Cyl(r,r=R3)\ Cyl(r,r=R4)] – "Zone:C3".
5. Down-camera – [Cyl(r,r=R2)\ Cyl(r,r=R3)] – "Zone:C2".
6. Steel vessel - [Cyl(r,r=R1)\ Cyl(r,r=R2)] – "Zone:C1".

All the system is in a vacuum.

For a comparative analysis of different functionals the reactor is subdivided into a set of registered objects (R_0, see Fig.C-4). Numeration of registered objects is presented in Fig. C-5 and C-6. It should be noted that every registered object represents a regular structure of smaller cells (see Fig. B-5). Besides every registered object is subdivided into registered areas 1-8 (see Fig. B-8).

Below 5 different core loadings are considered and corresponding patterns are presented in Fig. C-1, C-7, C-8, C-9 and C-11. In Table 3 a general list of calculational variants is presented; they differ by a core loading pattern and by the parameters defining a core state (see Table C-2). One of the parameters defining a core state is a position of CR (inserted or extracted from a core).

For the description of FA inner structure (location of different materials) the same principals as in Annex B are used (see Table B-3). So in Table c-1 a coordination is defined between FA type (type 1 or type 2) on a pattern with a FA type and the data in Table B-3 (K1-K6) taking into account a material name. A material name is formed using an initial name from Table B-3 by adding a material subtype (see Fig. C8, C-9, C-11). For example if in Table B-3 the material U-4.2 is defined and if its subtype (average burnup) is equal to 15 then in Table B-3 it is needed to use the material U_4.2:15 instead of U_4.2. Isotopic compositions of different materials are presented in Table B-1.

Табл. С-1 Описание группы тестов ВВЭР-1000

Номер системы	***1***	***2***	***3***	***4***	***5***
Название системы	***C U3.7*** (Fig. C-7, C-10)	***C PU3.6*** (Fig. C-7, C-10)	***C MIX B*** (Fig. C-8, C-10)	***C MIX E*** (Fig. C-9, C-10)	***C MIX 3*** (Fig. C-11, C-10)
Object-L1000	***K1 - 1*** ***(см.Табл.В-3*** ***Fig.C-5,C-1*** ***Fig. B-1***	***K2 - 1*** ***(см.Табл.В-3*** ***Fig.C-5,C-1*** ***Fig. B-1***	***K5 - 1, K6 - 2*** ***(см.Табл.В-3*** ***Fig.C-5,C-2*** ***Fig. B-2, B-3***	***K5 - 1, K6 - 2*** ***(см.Табл.В-3*** ***Fig.C-5,C-2*** ***Fig. B-2, B-3***	***K5 - 1, K6 - 2*** ***(см.Табл.В-3*** ***Fig.C-5,C-2*** ***Fig. B-2, B-3***
W_gap	Fig. C-2	Fig. C-2	Fig. C-2	Fig. C-2	Fig. C-2
HoleV	Fig. C-1	Fig. C-1	Fig. C-1	Fig. C-1	Fig. C-1
V	Fig. C-1 Mat-CL2	Fig. C-1 Mat-CL2	Fig. C-1 Mat-CL2	Fig. C-1 Mat-CL2	Fig. C-1 Mat-CL2
C3	Fig. C-1 Mat-CL2	Fig. C-1 Mat-CL2	Fig. C-1 Mat-CL2	Fig. C-1 Mat-CL2	Fig. C-1 Mat-CL2
C2	Fig. C-1	Fig. C-1	Fig. C-1	Fig. C-1	Fig. C-1
C1	Fig. C-1 Mat-CL2	Fig. C-1 Mat-CL2	Fig. C-1 Mat-CL2	Fig. C-1 Mat-CL2	Fig. C-1 Mat-CL2

Table C-2. Description of states

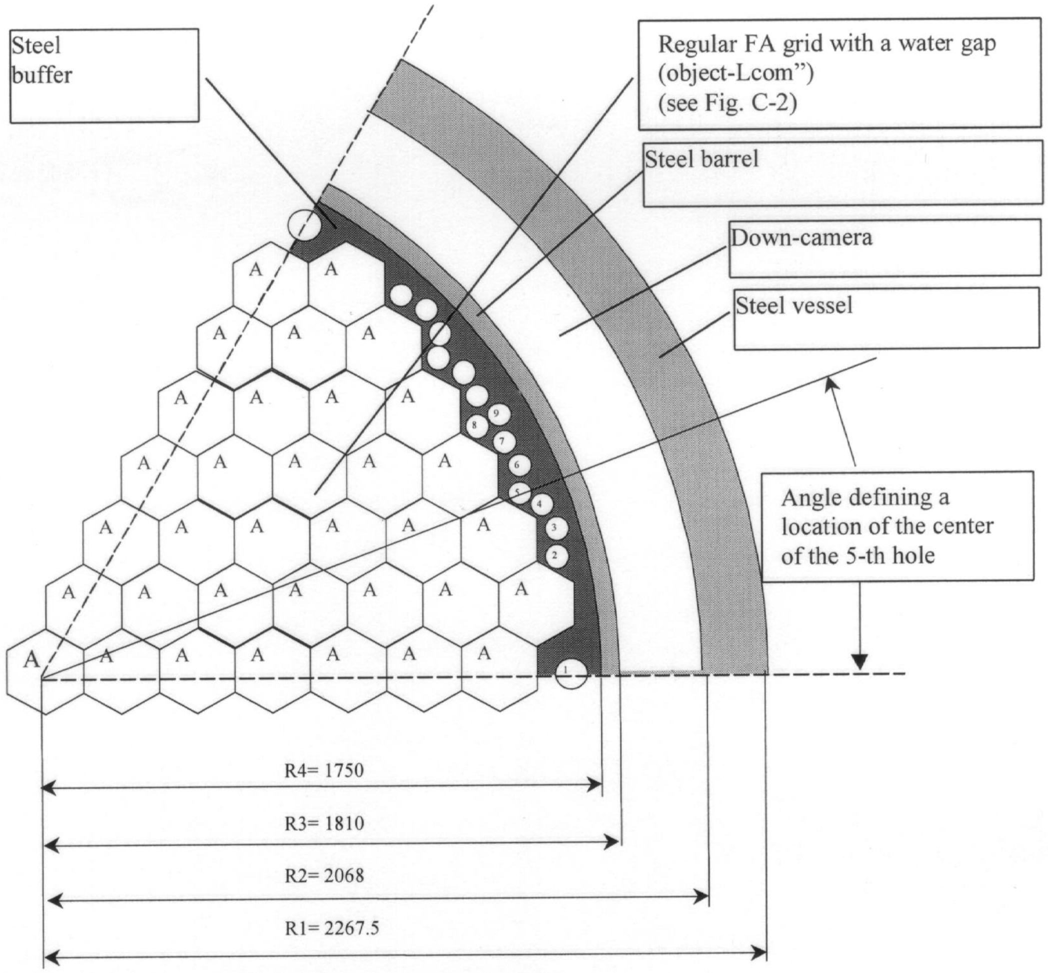
State Number	*State Name*	Zone: V,C1,C3 Temp.(K)	Zone: W_gap, HoleV,C2 Temp.(K)/mater.	Type: *Fuel* temper. (K)	Type: *Non_Fuel_zones* (temp., K)	Type: *Moderator_zones*	***Assembly:*: Into Guide tube (Fig. C-10)***	***Assembly:**: Into Guide tube (Fig. C-10)***	***Absorber zones:***
1	B1	560	560 / M560B1.3	1027	575	M575B1.3	***Mod_cell***	***Mod_cell***	-
2	B2	560	560 / M560B1.3	575	575	M575B1.3	***Mod_cell***	***Mod_cell***	-
3	B4	300	300 / M300B2.8	300	300	M300B2.8	***Mod_cell***	***Mod_cell***	-
4	E1	560	560 / M560B0	1027	575	M575B0	***Mod_cell***	***Mod_cell***	-
5	E2	560	560 / M560B0	575	575	M575B0	***Mod_cell***	***Mod_cell***	-
6	E3	553	553 / M553B0	553	553	M553B0	***CRcell***	***CRcell***	***B_4C_enr-***
7	E4	553	553 / M553B0	553	553	M553B0	***CRcell***	***Mod_cell***	***B_4C_enr-***

All calculations should be performed with zero axial leakage

Table C-3. List of calculational variants

Variant name Variant number	System number	State number/Xe-poisoning
1	1	1 (full power B=1300)
2	2	1 (full power B=1300)
3	3	1 (full power B=1300)/absent
4	3	2 (hot B=1300) /absent
5	3	5 (hot B=0) /absent
6	3	3 (cold B=2800) /absent
7	4	4 (full power B=0)
8	4	5 (hot, B=0)
9	4	6 (MCL, all CRs↓)
10	4	7 (MCL, all CRs↓ one ↑)
11-17	5	1-7

Fig. C-1. Location of different elements in VVER-1000 core model
(60° fragment, rotating symmetry)



Hole number	Distance from core center (R)	Angle	Hole diameter
	mm		mm
1	1655	0	98
2	1657.494	13.45506	70
3	1679.758	16.32916	70
4	1661.535	19.21195	70
5	1606.299	21.55143	70
6	1640.091	24.36647	70
7	1633.891	27.36905	70
8	1588.868	30	70
9	1675.47	30	70

Fig. C-2. Regular FA grid with a water gap
Object -"Lcom"
(60° fragment, rotating symmetry)

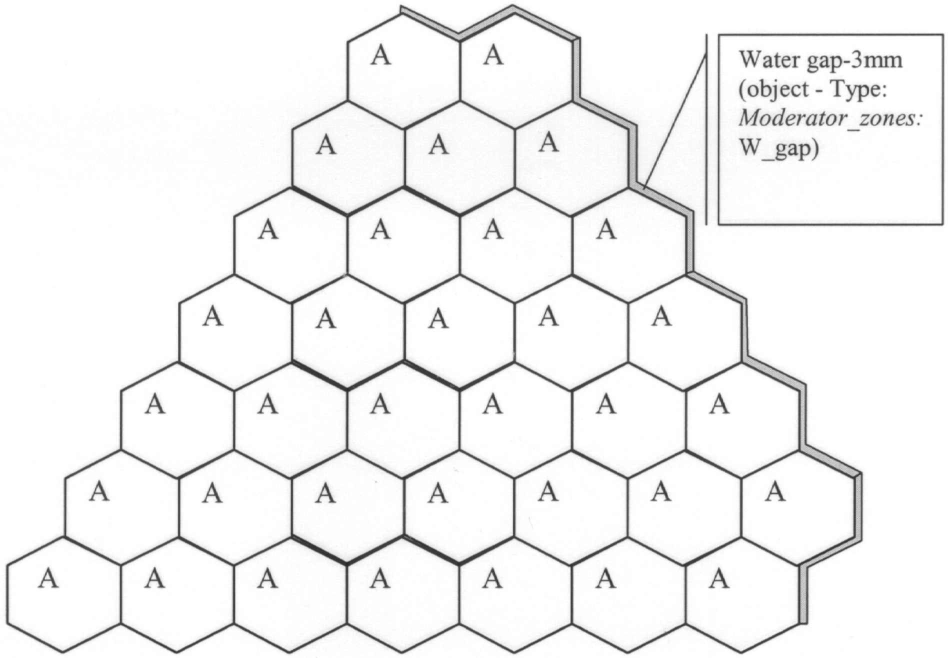


Fig. C-3. Regular FA grid with a water gap
Object -"L1000"
(60° fragment, rotating symmetry)

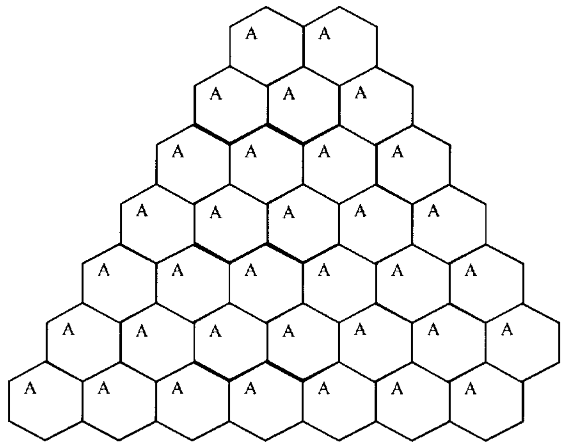


Fig. C-4. Location of the registration set of FA type objects in the system (60° fragment, rotating symmetry)

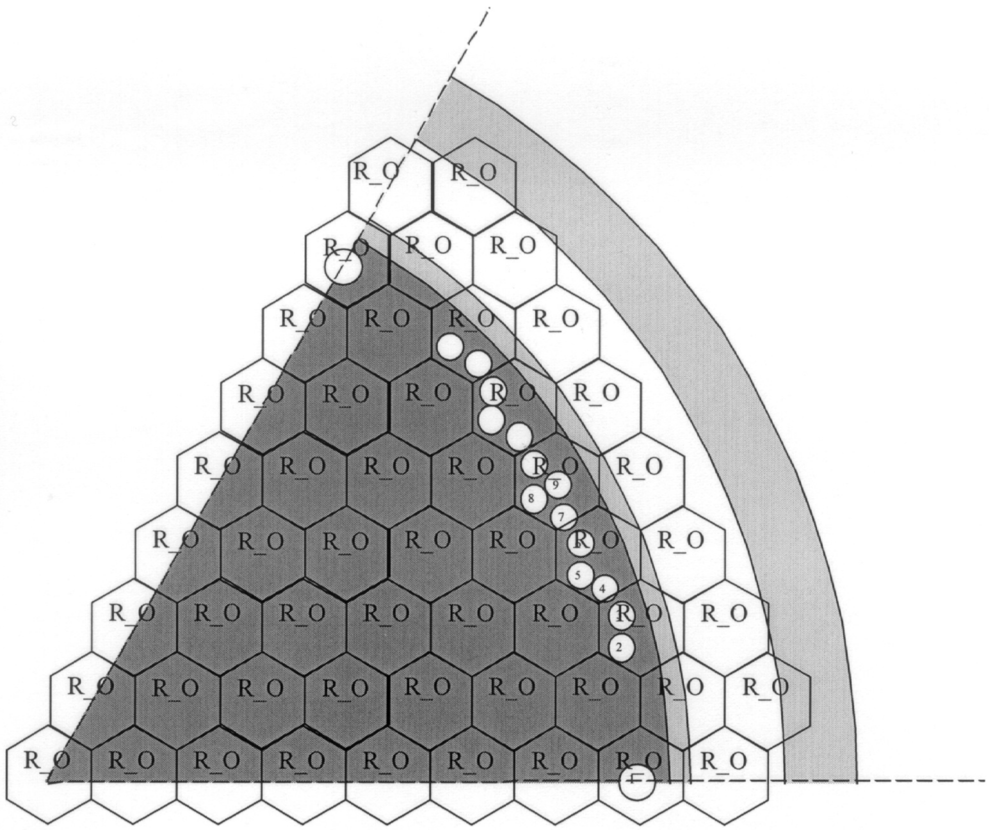


Fig. C-5. Numeration of the objects in the regulating set in 60° fragment
(See Fig. C-4, object - R_O)

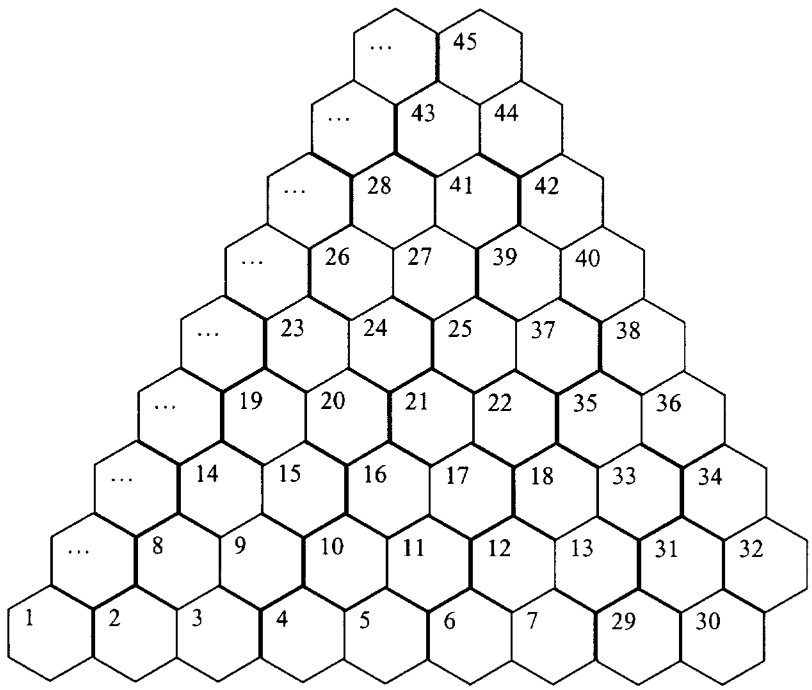


Fig. C-6. Numeration of the objects in the regulating set in 360° fragment
(See Fig. C-4, object - R_O)

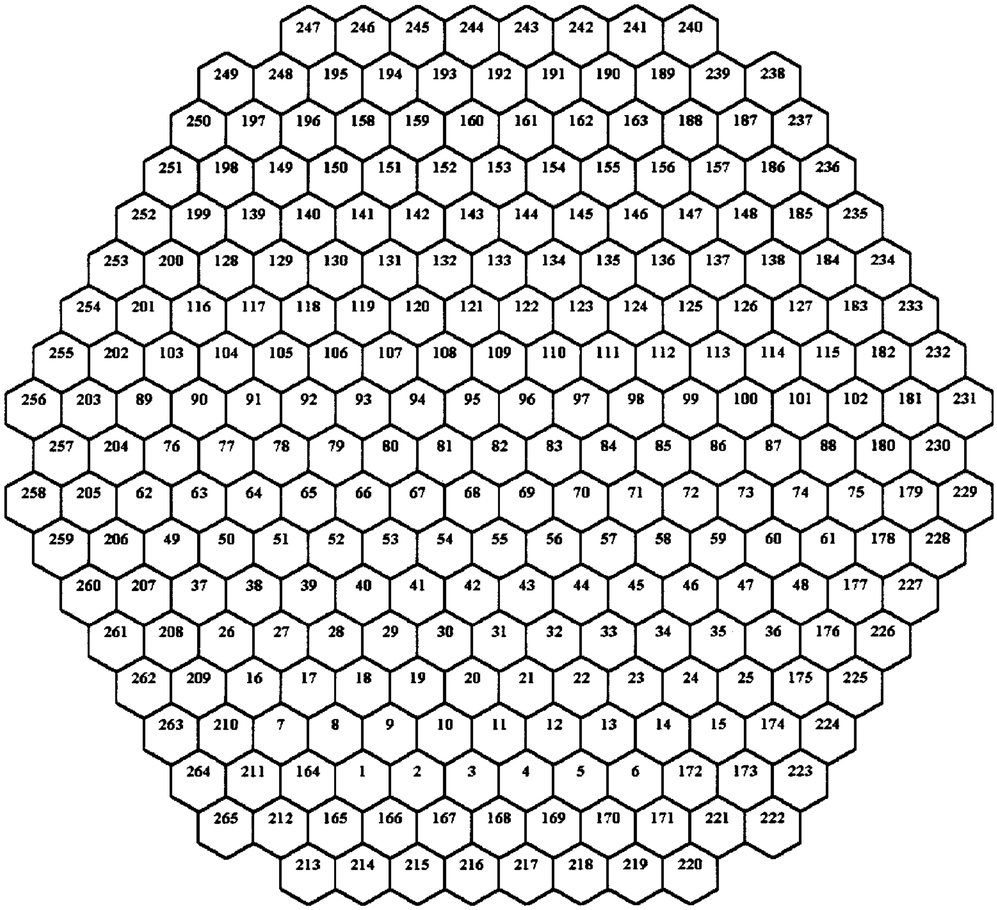
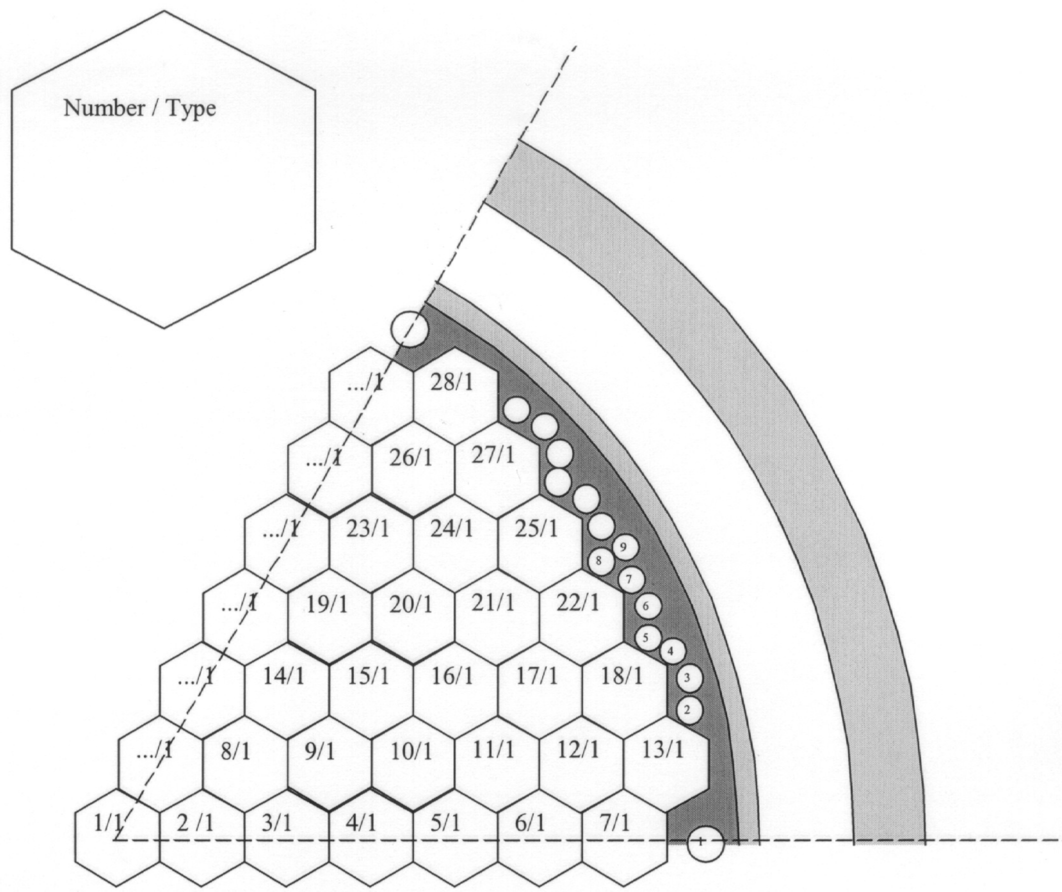
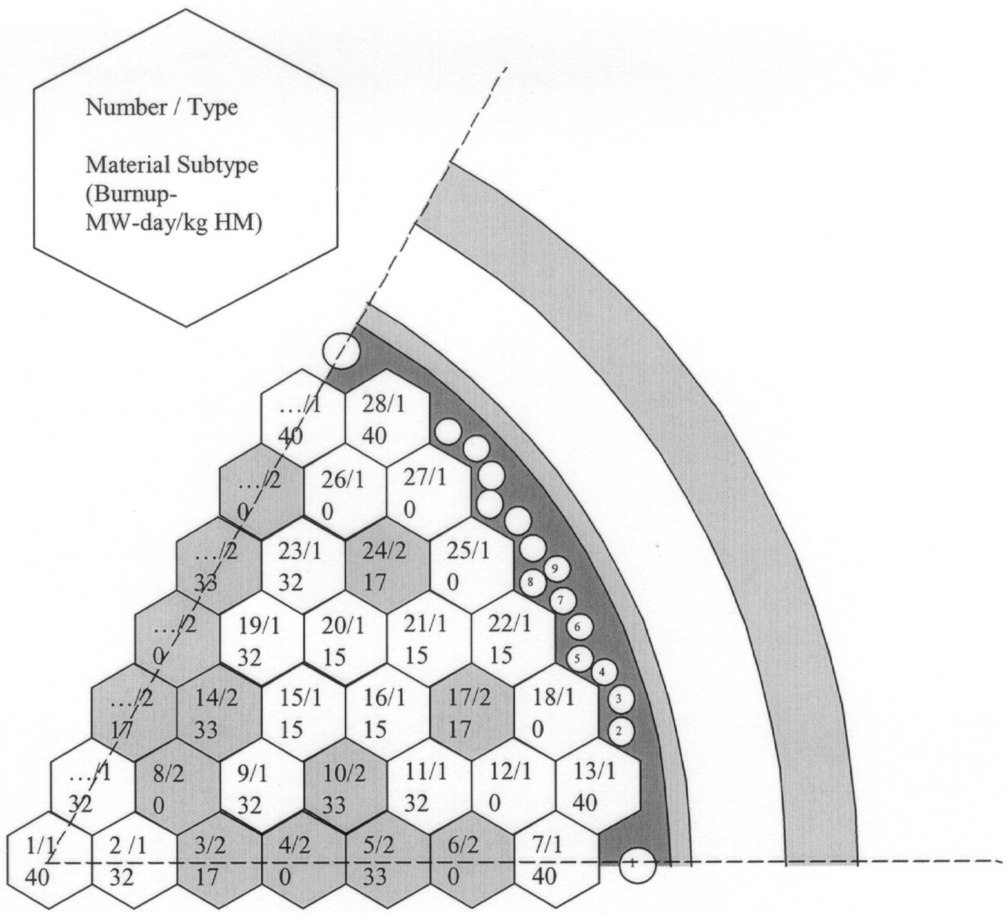


Fig. C-7 Pattern of FA types in a homogeneous loading
(60° fragment, rotating symmetry)



C-8. Pattern of FA types for BOC (30% MOX-fuel loading)
(60° fragment, rotating symmetry)



C-9. Pattern of FA types for EOC (30% MOX-fuel loading)
(60° fragment, rotating symmetry)

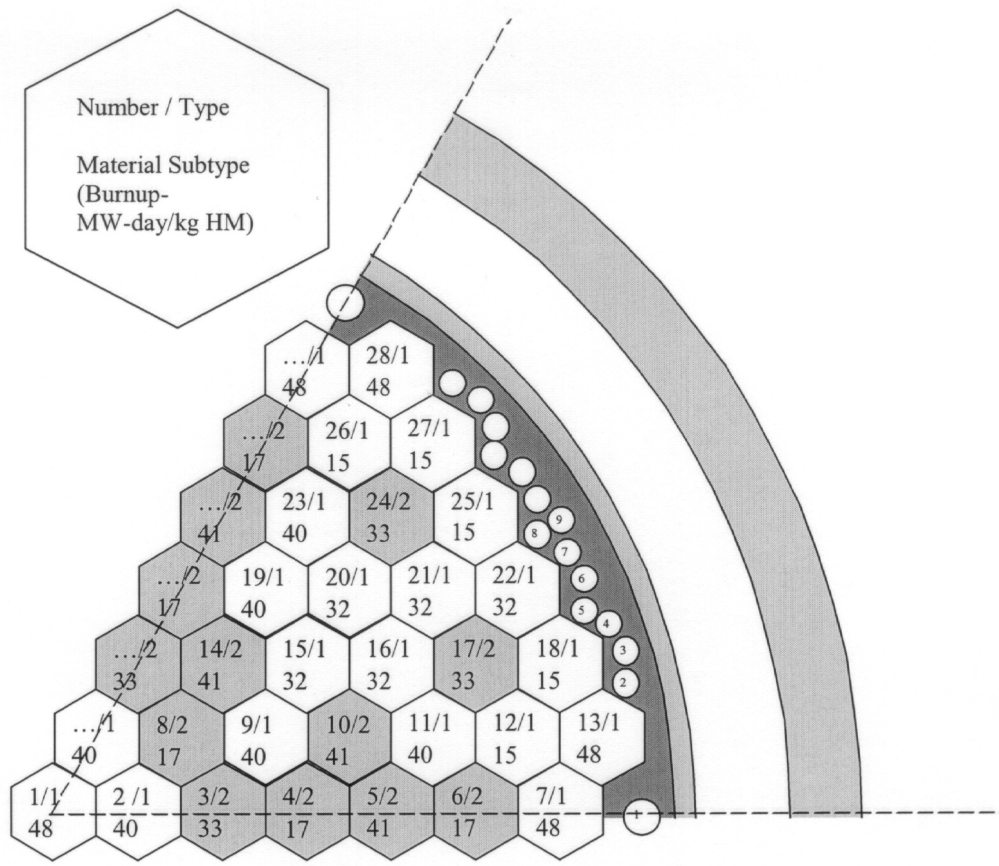
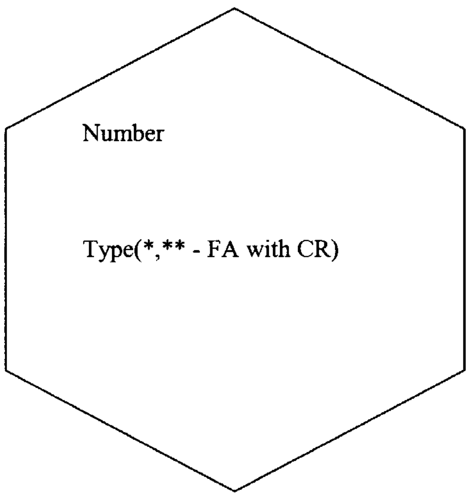


Fig.C-10. Numeration of FAs in a reactor core
(See Fig. C-1, C-2, C-3 , object-"L1000", 360° sector)



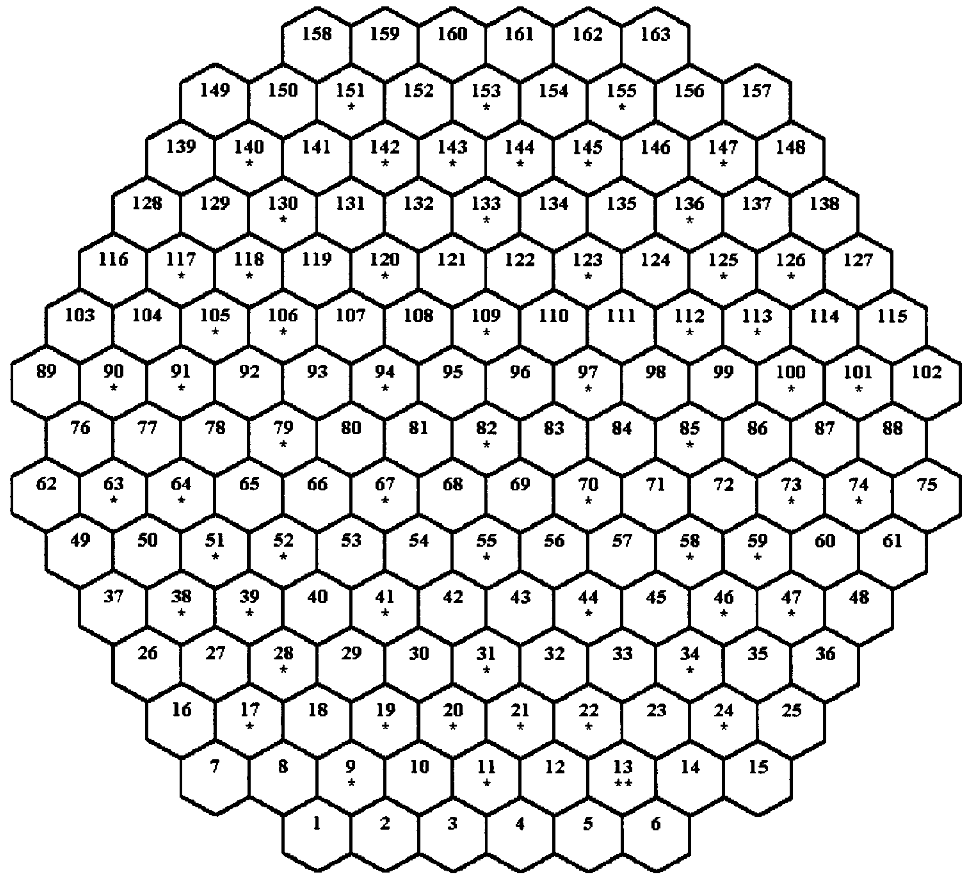
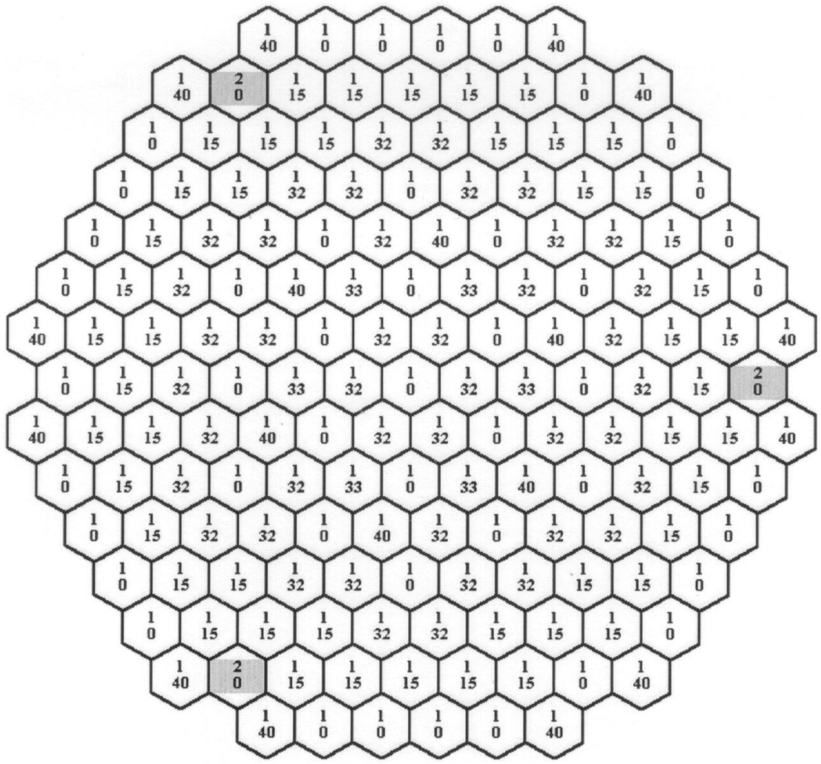
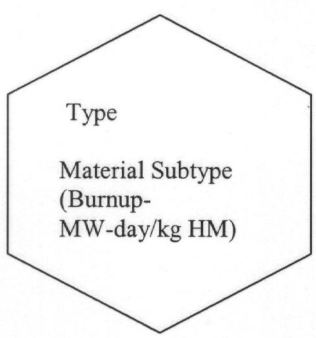


Fig. C-11 Pattern of the loading with 3 MOX LTAs





Comments from J. C. Gehin and R. T. Primm III, ORNL, on *Core Benchmarks Description Report*

TECHNICAL COMMENTS

1. In the sections covering the values to be reported, two-group neutron cross sections are requested. However, no breakpoint energy is provided. A value of 0.625 eV, which is standard, is assumed.

2. The geometric description on pages 43–46 is complicated. It could be more easily understood by Americans if it were supplemented by additional tables, figures, or a combination of both.

3. The material for the down-comer (not down-camera!) region is not specified in Table C-1. It is assumed to be the same at the core coolant for a given state.

4. “Down-camera” on page 57, in Table C-1, and on Fig. C-1 should be “down-comer.”

5. In Table C-1 the following system names should be defined: V (steel buffer), C3 (steel barrel), C2 (down-comer), and C1 (steel vessel). The corresponding letters should also be given in Fig. C-1. I do note that these are explicitly defined in the main body of the report (page 15).

6. The table of Acronyms is a little confusing. Many of the Russian acronyms are the same as their western equivalent. I would assume that there would be actual Russian acronyms for each of these.

7. For the minicore cases, why not deplete to a higher burnup than 24 MWd/kg? The number of time steps in the 0–10 MWd/kg could be reduced so that the number of burnup steps is not significantly increased. I do agree that most of the interesting aspects of the problem occur in the initial burnups. However, having high burnups would cover the entire expected range for the actual assemblies.

8. In Annex A, page 21, numbered item 2: It is assumed that “BIPR-8” should be “PERMAK.”

9. In Annex A, the last bullet on page 21: The phrase “serial Fa geometry” is taken to mean a “production-type” assembly. That is, an assembly design that has been approved for industrial fabrication; not a prototype or experimental design.

FORMATTING AND GRAMMATICAL COMMENTS

1. Generally well written but additional improvement is possible. The following comments do not affect the technical content of the report and are intended to provide feedback so that future reports may conform more to Oak Ridge National Laboratory standards.

2. Regarding the cover page, it is not certain how to determine the authors, and in particular the lead author of the report. Should A. M. Pavlovitchev be considered the lead author to reference as “A. M. Pavlovitchev et al.,” or should the reference be “P. A. Bolbov et al.,”? In the United States, the month is included in the publication date.

3. Rather than have a summary, it is U.S. convention to have an abstract that provides a good overview of the report. It should be self-contained such that it can be distributed separately from the report.

4. The “Introduction” section should be numbered in the same manner as the other sections. The sections of Annex A should be numbered A.1, A.2, etc.

5. Use of English language and grammar has improved over that of previous reports, but it is still a little difficult to read in places such as page 10, end of first paragraph. The authors clearly intend to say that if there is disagreement with PERMAK and BIPR results, then new studies will have to be defined. However, if there is agreement, then BIPR will have been found to be acceptable for use, or, at a minimum, acceptable for further study. In either case, the results are “valuable” for both cases, not only the case in which the results from the two codes agree (as the report states).

6. Page 11, Section 2.1: The phrase “neutron migration squares” most likely should be “neutron migration area (M^2).

7. All figures and tables must be numbered, and a “List of Figures” and a “List of Tables” should be included with the report. The figures on pages 8 and 9 are not labeled with figure captions.

8. A few Russian words and conventions are still in the report. The use of “_” in the last sentence of the first paragraph on page 5; the use of “<< >>“ in the second bullet on page 6; and the use of “,” rather than “.” in numbers on pages 43–46. Generally, however, the authors have done a good job of using the U.S. conventions.

9. Page 5, numbered point 3: The word “embarrassed” probably should be “restricted.”

10. In the paragraph on page 7, “neutron transfer” should be “neutron transport.”

11. In some places “Кo” is used where in other places “Ko” is used (note use of Cyrillic К versus K).

12. In the references, items 3 and 4 should be indicated as “Personal Communication” because they are not published or generally available. In references 1 and 2 the copyright indication is not necessary, just include the date in parentheses. Also, the ISBN and ANS order number are not typically given in the reference to a report. For convenience you can include this information in parenthesis at the end of the reference if you wish. The “ANS International Topical…the Next Millennium” should be in italics.

13. Each Annex, which is usually called an “Appendix” in the United States, should have a cover page with the Annex/Appendix letter (letters are typically used, not numbers) and the title.

14. Table B-1: The superscript on the boron isotopes should precede the letter “B” rather than after the letter.

15. Page 41, footnote to Table B-2: The exponent of Avagadro’s number should be 24 rather than 23.

16. Page 58: Some of the table titles have not been translated to English.

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