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PRESENT STATUS OF THE NET IBK COMPUTER CODE PACKAGE FOR IN-CORE FUEL MANAGEMENT AND RELATED CORE PARAMETER CALCULATIONS

M.V. MATAUŠEK, N. MARINKOVIĆ, I. PETROVIĆ,
N. ZAVALJEVSKI, M. MILOŠEVIĆ
Boris Kidrič Institute of Nuclear Sciences,
Beograd, Yugoslavia

Abstract

At the Nuclear Engineering Department of the Boris Kidrič Institute of Nuclear Sciences (NET IBK) systematic work has been going on for quite a period of time with the aim to complete a computer code package for reactor core analysis and design. Nuclear data available on the basis of international cooperation and exchange are being collected, while the already formed nuclear data libraries are being updated. Own methods, algorithms and codes are being developed. Foreign codes, supplied from different code centers, are being modified for application on the available computers and for the own needs. Developed computer programs are being transformed into well documented codes and modules. Computing codes are being designed for connecting the independent modules into complex modular schemes.

This paper presents and discusses the current status of the NET IBK computer code package for nuclear analysis of power reactors and in-core fuel management. The standard scheme for reactor fuel burnup analysis comprises the WIMS code and several 2 D (RZ or XY) and 3 D (XYZ) codes for overall reactor core calculations and criticality search. They are coupled and modified to compute neutron flux, power density distribution and burnup taking into account spatial variations of temperature and xenon poisonning, as well as the reactivity changes due to xenon transients during the start-up and shut-down. Presently, codes for overall reactor calculations are based on finite difference solution of group diffusion equations. Efforts are being made to improve reactor cell and fuel assembly parameters calculations, and to develop advanced methods for solving diffusion equations. Also, an optimization model, based on coarse zonal discretization of a reactor core is being developed for optimal fuel loading pattern search.

The NET IBK computer code package has been extensively used to study advanced fuel utilization schemes in different types of power reactors, as well as for solving in-core fuel management problems of the own reasearch reactors. Particular attention has been paid to experimental verification of the calculational procedures. In this paper, a number of interesting results is presented and discussed.

1. INTRODUCTION

For a developing country buying its first nuclear power plant from a foreign supplier, disregarding the type and scope of the contract, there is still a number of activities which has to be performed by local stuff and domestic organizations. This particularly applies to the choice of the fuel cycle strategy and the choice of the type and the size of nuclear power plants, to the bid parameters specification and bid evaluation, the evaluation of safety analysis reports, as well as to the in-core fuel management activities. To cope with these problems, a group of experts, possessing adequate methods, algorithms and computer codes, is indispensable.

The present paper describes methods, algorithms and codes for nuclear analysis of power reactors and in-core fuel management, developed and used at the Nuclear Engineering Department of the Boris Kidric Institute of Nuclear Sciences (NET IBK). It presents and discusses the current status of the NET IBK computer code package, showing its different possibilities, as well as its inadequacies and insufficiences.

The NET IBK computer code package has been extensively used to study advanced fuel utilization schemes in different types of power reactors, as well as for solving in-core fuel management problems of the own reasearch reactors. Particular attention has also been paid to experimental verification of the calculational procedures. A number of interesting results is presented and discussed here.

2. NET IBK IN CORE FUEL MANAGEMENT CODE LIBRARY

For quite a period of time, systematic work has been going on at NET IBK with the aim to complete a computer code package for reactor core analysis and design [1]. Nuclear data available on the basis of international cooperation and exchange are being collected, while the already formed nuclear data libraries are being updated. Own methods, algorithms and codes are being developed. Foreign codes, supplied from different code centers, are being modified for appliecation on the available computers and for the own needs. Developed computer programs are being transformed into well documented codes and modules. Computer codes are being designed for connecting the independent modules into modular schemes of different degree of complexity and of different assignment.

Codes available at NET IBK for solving in-core fuel management problems and for caculating the related core physics parameters are listed in Table I. A brief description of their function, input, output, status and origin is given. Items 1 to 6 are the basic nuclear data libraries, programs marked by numbers 7 to 10 are used for cross sections and resonance integrals compilation and evaluation, while programs marked by numbers 11 to 20 are used for neutron spectrum calculation, pin cell and fuel assembly studies and generation of group constants needed for overall reactor core design. Items 21 to 23 are the static design codes used for calculating reactivity and flux and/or power distribution for a given reactor core configuration and a given set of few group parameters. Programs denoted by numbers 24 to 36 are used for fuel burnup analysis and in-core fuel management. They perform fuel depletion calculation, heavy isotope and fission product buildup and decay computation, while most of these codes are in fact complex modular schemes for calculating overall reactor core parameters versus burnup, i. e. for adjusting certain parameters to prescribed specification.

In the following paragraphs some of the programs listed in Table I will be described in more detail.

3. *STANDARD NET IBK SCHEME FUEL BURNUP ANALYSIS AND IN CORE FUEL* MANAGEMENT CALCULATION

Fuel burnup analysis comprises: neutronic calculations of fuel pin cell parameters versus burnup; neutronic calculations of few group parameters for fuel assemblies with inserted or withdrown control rods, versus burnup and for specified values of fuel, coolant and moderator temperatures; few group diffusion theory calculation of criticality parameters and neutron flux and/or power distribution in a reactor core with specified configuration and fuel burnup; calculation of related thermo-hydro-dynamic parameters. Repeating the global reactor core calculations with a certain burnup step, the entire range of fuel burnup values can be covered. The feedback between neutronic and thermo-hydro-dynamic calculations can be provided by introducing new temperatures in neutronic calculations when necessary.

The standard NET IBK calculational scheme for solving in-core fuel management problems consists of the WIMS code, for calculating burnup dependent pin cell and fuel assembly parameters, and of several 2 D (RZ or XY) or 3 D (XYZ) codes for overall reactor core calculations and criticality search. They are coupled and modified to compute neutron flux, power density distribution and burnup taking into account spatial variations of temperature and xenon poisonning, as well as the reactivity changes due to xenon transients during the start-up and shut-down. Presently, codes for overall reactor calculations are based on finite difference solution of group diffusion equations. Efforts are being made to improve reactor cell and fuel assembly parameters calculations, and to develop advanced methods for solving diffusion equations.

The NET IBK computer code package has been extensively used to solve in-core fuel management problems of the own reasearch reactors, as well as for studying advanced fuel utilization schemes in different types of power reactors [10]. In doing so, particular attention has been paid to experimental verification of the calculational procedures [11]. Some typical results, obtained in the frame of several studies performed for the utility organizations [12] will be presented here.

Table I List of NET IBK Codes for Solving In-core Fuel Management and Related Core Physics Problems

no.	Name	Program description	Input	Output	Status in NET	Origin	Reference
1.	UKNDL	Evaluated nuclear data file	-	-	In use	Obtained from IAEA	AWRE-O-70/63
2.	KEDAK	Evaluated nuclear data file	-	-	In use	Obtained from IAEA	KFK-750
3.	ENDF/B-IV	Evaluated nuclear data file	-	-	In use	Obtained from IAEA	IAEA-NDS-23
4.	BOND	Fast reactor group constants of Abagian-Bondarenko type	-	-	In use	Developed in NET	[2]
5.	JENDL-2	Evaluated nuclear data file	-	-	In use	Obtained from IAEA	IAEA-NDS-18
6.	SOKRATOR	USSR evaluated nuclear data library	-	-	In use	Obtained from IAEA	INDC(CCP)-97
7.	NJOY	Calculation of pointwise and multigroup cross sections from ENDF/B-IV and -V libraries	ENDF/B-IV	Pointwise and multigroup cross sections	Being tested	Obtained from RSIC	PSR-171
8.	PARMA	BOND library service routine	BOND data	Multigroup data for a given mixture of nuclides	In use	Developed in NET	IBK-1267
9.	SIGMA	Calculation of multigroup thermal data (free gas model)	UKNDL or KEDAK	Thermal multigroup data for given nuclides	Has been used*	Developed in NET	IBK-1282
10.	GRCON	Calculation of multigroup thermal data for water (Nelkin)	-	Thermal multigroup data for light or heavy water	Has been used	Developed in NET	IBK-1298
11.	MONO	Solution of monoenergetic transport equation in 1-D cylindrical geometry by spherical harmonics method	Results of SIGMA and/or GRCON	Radial distribution of neutron flux in a multi-zone reactor lattice cell	Has been used	Developed in NET	IBK-654
12.	MULTI	Solution of multigroup transport equation in 1-D cylindrical geometry by spherical harmonics	Results of SIGMA and/or GRCON	Space-energy distribution of neutron flux in a multi zone reactor lattice cell	In use	Developed in NET	IBK-1210
13.	THERMOS	Transport equation solution in 1-D cylindrical geometry by collision probability method	THERMOS library	Space-energy distribution of neutron flux in a multi zone reactor lattice cell	Has been used	Obtained from NDB	IBK-1411
14.	VINRAM	Monte Carlo criticality calculation of one dimensional spherical or cylindrical systems	VINRAM library	Criticality parametres, neutron spectrum	Has been used	Developed in NET	IBK-1459
15.	CIPET	Solution of multigroup transport equation in 2-D cyclindrical geometry by spherical harmonics	BOND library	Criticality parametres, neutron spectrum	In use	Developed in NET	[3]
16.	KASETA	Monte Carlo calculation of space and energy neutron flux distribution in a PWR assembly	Results of SIGMA and/or GRCON	Few group constants for overall reactor calculations	Has been used	Developed in NET	IBK-1377
17.	MC**2-2	Solution of neutron slowing down equation and generation of multigroup cross sections	ENDF/B data	slowing down spectra and multigroup cross sections	Not tested	Obtained from NDB	ANL-8144
18.	SPLET	Solution od slowing down transport equation in 1-D cylindrical geometry by spherical harmonics method	UKNDL or KEDAK	Space-energy distribution of epithermal neutron flux in a multizone cell	In use	Developed in NET	[4], IBK-1055
19.	PARTI	Group collapsing based on an optimizational procedure	Detailed spectra	Optimal group structure	In use	Obtained from NDB	IRW-4332
20.	VESNA	Multigroup pin cell and fuel assembly calculation	VESNA library	Criticaity parameters and few group diffusion constants	In use	Developed in NET	[5]

Table I - continued

no.	Name	Program description	Input	Output	Status in NET	Origin	Reference
21.	2D-6RAND	Solution of 2-D few group diffusion equation by finite difference method	Few group data	Space distribution of group neutron fluxes and criticality parameters	Has been used	Obtained from NDB	ORNL-3200 IBK-1374
22.	KAK	Solution of 2-D few group diffusion equation by finite difference method	Few group data	Space distribution of group neutron fluxes and criticality parameters	Has been used	Obtained from NDB	WCAP-3269-26
23.	KONEL	Solution of 2-D few group diffusion equation by finite element method	data from WIMSD-4	Criticality parameters and neutron flux space distribution	In use	Developed in NET	[6]
24.	CINDER	One point depletion and fission product calculation	Results from MULTI, SPLET	Fissile nuclides and fission products number densities	In use	Obtained from NDB	WAPD-TM-334 IBK-1414
25.	LASER	Depletion calculation for a lattice cell	LASER library	Burnup dependent number densities and cell parameters	In use	Obtained from ANL	WCAP-8073
26.	LEOPARD	Cell homogenization and spectrum generation with fuel depletion	LEOPARD library	Burnup dependent number densities and cell parameters	In use	Obtained from ANL	WCAP-3269-26
27.	WIMSD-4	Multigroup pin cell and fuel assembly calculation and generation of few group parameters versus burnup	WIMS library	Few group data for overall reactor calculations, criticality parameters	In use	Obtained from NDB	AEEW W 1327 IBK-1576
28.	IZGOR	Reactor core burnup analysis and criticalyty search by solving 2-D few group diffusion equation (modification of 2D-6RAND)	Data from WIMSD-4	Space distribution of group neutron fluxes and criticality parameters	In use	Developed in NET	[7]
29.	EREBUS	Reactor core burnup analysis and criticalyty search by solving 2-D multigroup diffusion equation (finite differences)	microscopic multigroup data	Space distribution of neutron flux and power density and criticality parameters versus fuel burnup	In use	Obtained from NDB	FN-E-88
30.	VAMPIR	Reactor core burnup analysis and criticalyty search by solving 2-D multigroup diffusion equation (modification of EREBUS)	data from WIMSD-4	Criticality parameters and power space distribution	In use	Developed in NET	[8]
31.	TRITON	Reactor core burnup analysis and criticalyty search by solving 3-D multigroup diffusion equation	microscopic multigroup data	Power and neutron flux space distribution and criticality parameters	In use	Obtained from NDB	FN-E-97
32.	CITATION	Solution of 1-D, 2-D or 3-D few group diffusion equation for fuel management purposes	few group data	Power and group fluxes space distribution and criticality parameters	Being tested	Obtained from ANL	ORNL-TM-2496
33.	MARIA	Code block for PWR fuel assembly calculations	WIMS library	Few group reactor core parameters versus burnup	Not tested	Obtained from NDB	JEN 543
34.	LOLA	3-D, one group PWR fuel management calculations by nodal method	MARIA data	Burnup dependent core parameters with criticality search and control rods insertion	Not tested	Obtained from NDB	JEN 568
35.	JOSHUA	2-D, 3-D neutronics, hydraulics, burnup, refuelling for LWRs	WIMS data	Overall power distribution and reactivity parameters	Not tested	Obtained from NDB	AEEW-R1186
35.	CICLON	Neutronics calculations for PWR transition cycles	WIMS data	Cycle length, fuel enrichment for given load factor	Has been used	Obtained from NDB	NEA 451
36.	SNOOPY	Optimization of fuel loading by coarse zonal discretization model		Optimal fuel loading pattern	In use	Developed in NET	[9]

* "Has been used" means that the program was written and tested on the computer CDC 3600 which is no longer in use and has not been used on presently available computers

In order to get elements for evaluating the overall fuel cycle strategy, a comparative study of burnup parameters for different types of 1000 MWe nuclear power plants has been performed. For this kind of study, it is appropriate to consider the results of pin cell calculations. The basic input data concerning the standard fuel management schemes in present day nuclear power plants, provided by different suppliers, are presented in Table II. In Table III the calculated values of heavy isotope compositions of fuel discharged from different types of nuclear power plants are compared with the reference calues. Satisfactory agreement can be noticed.

Nuclear power plants expected to be built in Yugoslavia in the future will probably be of an advanced type, with prolonged fuel cycles leading to discharge burnup values considerably higher than the ones achieved with the present day plants. In treating the effects of prolonged fuel burnup, changes in the design of fuel assemblies, core configuration and reactivity control system, introduced in order to provide longer fuel irradiation, are to be considered. Here, the specific design characteristics, as increased initial enrichment, advanced type burnuble absorbers and low-leakage loading patterns, are taken as for a typical advanced 1000 MWe PWR's offered by potential suppliers.

Table II - Data used in pin cell fuel burnup calculations

type of 1000 MWe NPP	PWR	VVER	BWR	PHWR
first core initial enrichment (kg ^{235}U/MTU)	25	24	20	7.11
equilibrium core initial enrichment (kg ^{235}U/MTU)	33	44	7.11	25.6
average discharge fuel burnup (GWD/MTU)	33	40	28.5	7.5
average power density in fuel (kW/kgU)	38.3	45.5	24.2	19.1
fuel pellet radius (mm)	8.19	7.55	10.6	14.2
thickness of Zr cladding (mm)	0.57	0.65	0.66	0.38

Table III - Calculated and reference values of heavy isotope compositions in fuel discharged from different types of nuclear power plants (kg/MTU)

isotope	235 U	236 U	238 U	239 Pu	240 Pu	241 Pu	242 Pu
					PWR		
calculated	8.376	4.193	941.0	5.932	2.354	1.428	0.581
reference	9.000	3.800	940.7	5.407	2.379	1.299	0.473
					VVER		
calculated	12.575	5.188	930.3	5.497	2.379	1.644	0.654
reference	12.600	5.000	930.0	5.600	2.100	1.800	0.600
					BWR		
calculated	8.428	3.596	946.7	4.261	2.867	1.337	1.289
reference	8.400	3.500	947.0	4.576	2.056	0.928	0.360
					HWR		
calculated	2.112	0.743	985.6	2.665	0.878	0.211	0.588
reference	2.270	0.733	985.5	2.599	0.973	0.174	0.530

The "cluster" option of the WIMSD-4 code has been used to calculate group values of the efective physical parameters versus burnup for a fuel assembly with 17x17 positions in a square lattice. The portion of the fuel assembly assigned to one control rod, i.e. its 1/24 part is simulated by a multizone cylindrically symmetrical lattice cell. Calculations have been performed for an equilibrium cycle with the fresh fuel initial enrichment being 3.35% or 3.55% of ^{235}U. The following cases have been treated. fuel assembly without control rods or burnable absorbers, fuel assembly with integral type burnuble absorbers (a thin boride coating applied to the surface of the fuel pellets, i.e. combination of fuel and burnuble absorber material in a single rod) and fuel assembly with inserted control absorber rods. Infinite medium criticality parameters versus burnup for the first two cases are presented in Table IV.

The reactor core burnup analysis has been performed using the program VAMPIR [8], which solves few group difusion equations for a multizone reactor in twodimensional R-Z or X-Y geometry by the finite difference

Table IV - Infinite medium criticality parameters versus burnup for a fuel assembly with and without integral type burnable absorber

burnup (MWD/MTU)	with burnable absorber		no burnable absorber	
	k-inf	reactivity (pcm)	k-inf	reactivity (pcm)
0	.94765	-5523.75	1.36586	26786.05
150	.93415	-7049.61	1.31632	24030.69
4000	1.06837	6399.12	1.26539	20973.17
8000	1.12315	10965.01	1.21606	17766.88
12000	1.13187	11651.02	1.17163	14648.82
16000	1.11748	10513.18	1.13151	11622.21
22000	1.07833	7264.01	1.07721	7167.76
28000	1.03285	3180.33	1.02732	2659.35
34000	.98740	-1275.69	0.98079	-1959.06
40000	.94439	-5888.98	0. 3778	-6635.06

method. As input data, the program uses precalculated tables of group parameters versus burnup, for different values of the reactor material temperatures and densities. The program calculates spatial distributions of neutron flux, power density and fuel burnup as functions of reactor operating time. Two options are available: (1) calculation of criticality parameters versus full power operating time or (2) calculation of control rods positions providing the given criticality conditions.

Effects of prolonged fuel burnup have been studied by considering fuel management scheme based on four 12 month cycles. At the beginning of cycle (BOC) the equilibrium core thus contains 1/4, i.e. 48 fresh fuel elements, besides the once, twice and three times burned fuel. Assemblywise power and burnup distributions for hot and full power core with no control rods or burnuble absorbers, at BOC of an equilibrium cycle, have been calculated by the R-Z option of the program VAMPIR. The fresh fuel enrichment was 3.35% ^{235}U. The results given in Fig.1. show that in this case the maximum value of power form factor, i.e. assembly power/core averaged power, is very high (1.94).

One of the possible ways to perform power flattening is by insertion of control absorber rods. Here it is supposed that control rods are grouped in three banks moving independently. Using the option (2) of VAMPIR program the position of control banks is determined which considerably flattens power generation in the reactor core. The results are presented in Fig.2., where X1, X2, and X3 denote control banks unserted up to 73.134 cm, 10.445 cm and 20.895 cm from the bottom of the reactor

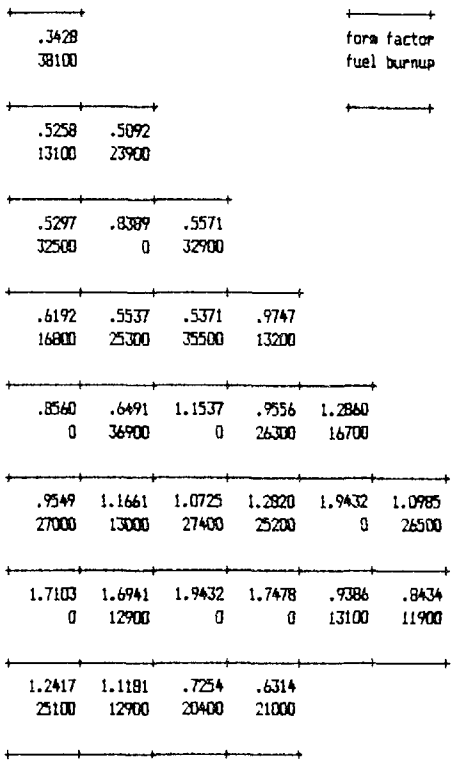

Figure 1. Assemblywise distribution of power form factors and burnup (MWD/MTU) for hot and full power core with no control rods or burnable absorbers, BOL of an equilibrium cycle

vessel. Comparing to the previous case the range of the power form factors is considerably decreased (0.67 - 1.46 instead of 0.34 - 1.94).

In a PWR core control rods are primarily used to meet rapid transient reactivity requirements, as well as safety shutdown requirements. When the fresh fuel initial enrichment is increased in order to provide prolonged fuel burnup, besides boron dissolved in the reactor coolant, burnable absorbers have to be used to control excess reactivity and avoid strong power peaking at the beginning of a cycle. The so called integral type of burnuble absorber, i.e. the combination of fuel and burnuble absorber material in a single rod, is considered here. About 80% of fuel pellets in the central part of a fuel rod are supposed to have thin (less than 2.5×10^{-3} cm) boride coating, while top and bottom portion of the fuel pellets are supposed to be uncoated in order to reduce axial peaking

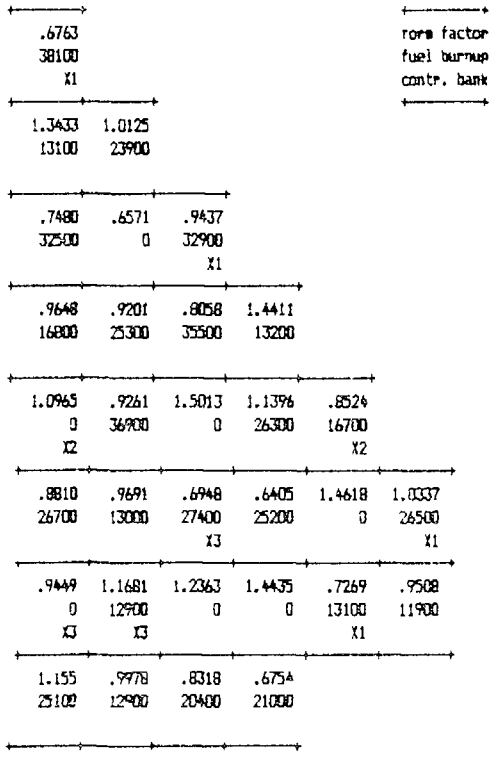

Figure 2. Assemblywise distribution of power form factors and burnup (MWD/MTU) for hot and full power core with inserted control rods, no burnable absorbers, BOL of an equilibrium cycle

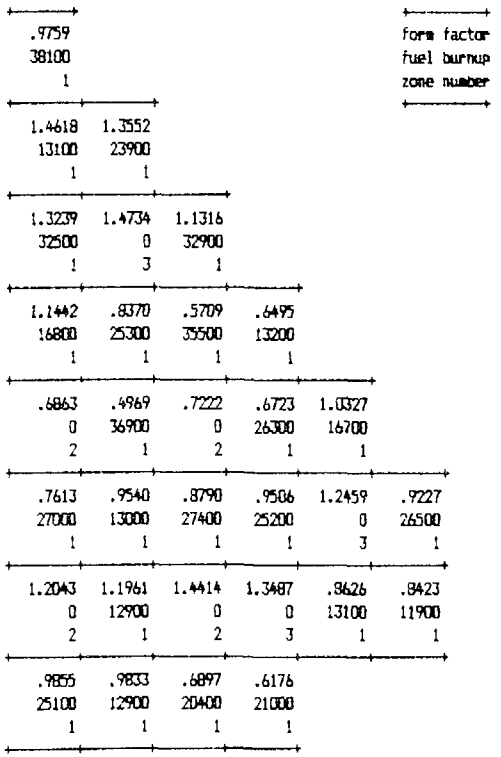

Figure 3. Assemblywise distribution of power form factors and burnup (MWD/MTU) for hot and full power core with integral type burnable absorbers, BOL of an equilibrium cycle

factos. In Fig. 3. calculated values of assemblywise distribution of power form factors and fuel burnup are presented for hot and full power core at BOC of an equilibrium cycle. The core configuration is supposed to consist of 145 fuel assemblies having initial enrichment 3,35% ^{235}U (zone 1), 24 fuel assemblies having initial enrichment 3,55% ^{235}U and integral type burnuble absorber (zone 2) and 24 fuel assemblies having initial enrichment 3,55% ^{235}U and no burnuble absorber (zone 3). In this case the range of power form factors is 0,61 - 1,46, i.e. approximately the same power flattening is achieved as in the case when groups of control rods nave been used.

During the fuel irradiation burnable porison is depleted, so that at the end of cycle its influence at the power distribution is practically eliminated. In spite of the fact that fuel depletion itself tends to flatten the flux and power distribution, power peaking can again become a problem at the cycle end. It can be seen in Fig. 4. that for the case treated here the range of power form factors at end of cycle (EOC) is 0,2 - 1,86, i.e. almost as in the case presented in Fig.1.

In Table V criticality parameters versus burnup for the equilibrium core with lower fresh fuel enrichment and no burnuble absorbers are compared with criticality parameters of the equilibrium core with higher fresh fuel enrichment and burnuble absorbers. In the later case about 9% longer cycle, i.e. about 4.5% higher average discharge burnup, can be achieved. Redistribution of zone averaged power and zone averaged fuel burnup during an equilibrium cycle containing burnuble absorbers is presented in Table VI. Figs. 5. and 6. show axial distribution of assembly power for different cases treted.

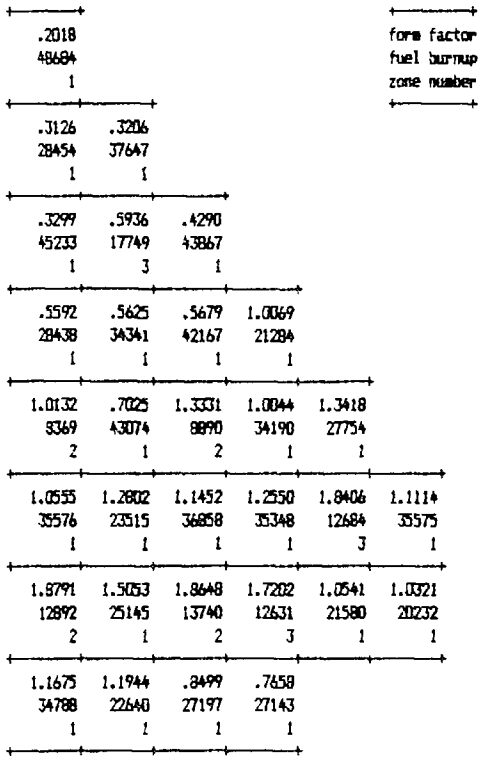

Figure 4. Assemblywise distribution of power form factors and burnup (MWD/MTU) for hot and full power core with integral type burnable absorbers, EOL of an equilibrium cycle

The purpose of the work presented here was to analyse the improved fuel designs and advanced in-core fuel management schemes in order to get elements for evaluating bids for future nuclear power plants, as well as to verify the available methodology and codes for burnup prediction. It can be concluded that different improvements in fuel design and fuel management, introduced in order to get prolonged fuel burnup and better fuel cycle economy, require more sophisticated calculational models and codes. Faster codes which make possible more detailed and numerous calculations in the frame of different optimization procedures are particularly desirable.

Table V - Criticality parameters versus burnup for the equilibrium core with 1/4 (48) fresh fuel assemblies

enrichment 3.35% number of burnable absorbers 0			enrichment 3.55% number of burnable absorbers 24		
oper. time (days)	reactivity (pcm)	burnup (MWD/MTU) average	oper. time (days)	reactivity (pcm)	burnup (MWD/MTU) average
0	9537.4	16608	0	6420.3	16607
100	4756.0	20060	100	2763.1	19988
200	1479.5	23512	200	1741.1	23367
280	-1152.8	26273	300	-822.3	26742

Table VI - Redistribution of zone averaged power and zone averaged fuel burnup during the life of an equilibrium core containing burnable absorbers (1/2 of the core)

	operating time of the reactor (days)				
zone	0	4.2	100	200	300
	zone averaged power (GW)				
1	.9555	.9699	.9542	.9281	.9261
2	.1205	.1208	.2151	.2457	.2553
3	.3363	.3216	.2430	.2385	.2309
total	1.4123	1.4123	1.4123	1.4123	1.4123
	zone averaged burnup (MWD/MTU)				
1	22105	22237	25259	28363	31383
2		94	2230	6200	10734
3		261	5895	10342	14706
average	16607	16751	19987	23366	26741

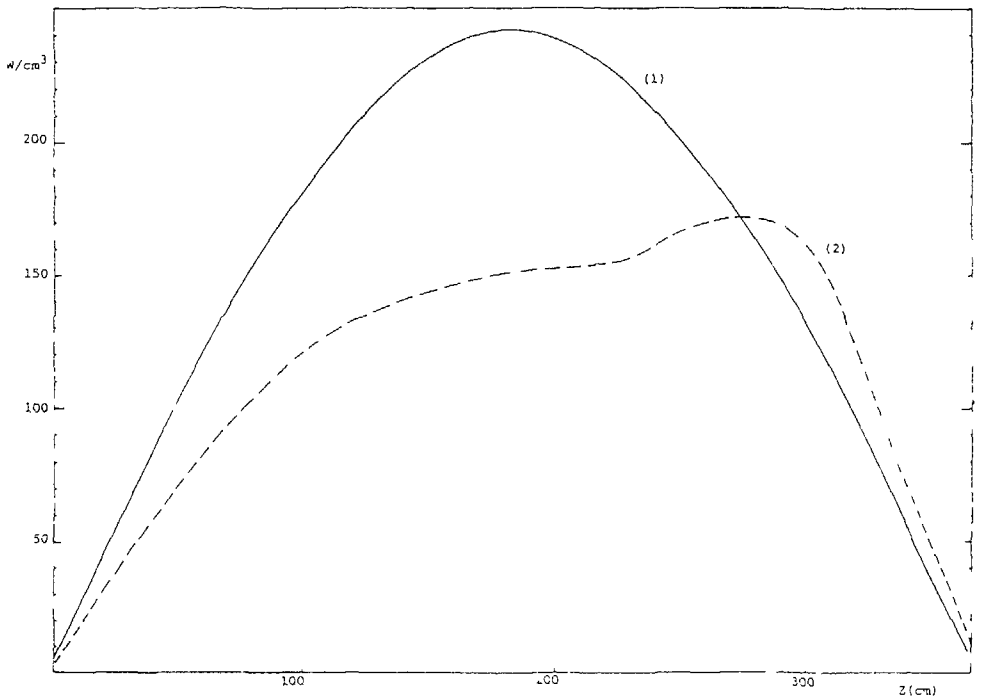

Figure 5. Hot channel axial distribution of power density HFP core, BOC of equilibrium cycle, no burnable poisons, (1) no control rods, (2) control rods inserted.

Efforts are thus being made to improve reactor cell and fuel assembly parameters calculations, and to develop advanced methods for solving diffusion equations. Also, an optimization model is being developed for optimal fuel loading pattern search.

4. CODES DEVELOPED AT NET IBK

Besides the codes obtained from different code centers, which are being modified for application on the available computers and for the own needs, several new NET IBK computer programs are being developed to be used either as alternative module in existing modular schemes or as independent codes.

A code called SPLET has been developed for solving an energy dependent one dimensional transport equation for a cylindricalized multizone reactor lattice cell. The method of solution is a combination of P-3 spherical harmonics method and a super fine energy discretization procedure [4]. The calculated detailed space energy distribution of epithermal enutrons in a reactor lattice cell is used to produce related integral quantities as reaction rates and resonance integrals, taking into account exact cross section resonance structure of all the materials present. The code is also meant to be combined with the program PARTI, which can calculate the optimal group structure for the given spectrum, in order to generate group cross section library particularly suited for certain reactor systems.

A multigroup two dimensional transport theory criticality code (IPET is based on the method proposed for solving 2-D multigroup P-3 spherical harmonics equations in R-Z geometry [3]. It calculates criticality parameters and space-energy distribution of neutrons in either a reactor lattice cell (a source problem), or a reactor as a whole (an eigenvalue problem). The code is suitable for producing reference results for particular cases. Presently, it will be used for obtaining accurate fluxes near the core boundary in order to study neutron leakage problems.

A general multigroup code called VESNA [5] has been developed for reactor lattice cell calculation and nuclear fuel assembly design for a wide range of reactor systems (PWR, BWR, PHWR, VVER). Although it belongs to a class of codes, all similar in structure and strategy, which may be characterized by the spectrum and spatial calculations being performed in 2D and in a single job step for the entire assembly, many important improvements have been added in the code VESNA, which significantly influence the performance of the resulting computational tool.

In both micro-group spectrum and macro-group assembly calculations, the VESNA code uses identical methods for the solution of the integral transport equations in 1D and 2D geometries, based on the ray tracing techniques for generation of collision probability matrices. Leakage and elasic scattering anisotropy are treated in B1 aproximation. The primary library is in 44 energy groups. The epithermal range down to 1 eV has 22 energy groups of which 13 are in the ^{238}U resonance range. Multigroup cross section data have been obtained from the BNAB-78 and ENDF libraries. Multigroup values of the selfshielding factors, calculated by the fine-multigroup slowing-down code, are tabulated for several temperatures and effective scattering cross sections. Group-by-group geometry dependent effective scattering cross sections are obtained by equivalence theory and IR approximation. Resonance shielding factors for ^{235}U and ^{239}Pu are given by tables of two quantities depending on $^{238}N/^{235}N$ or $^{238}N/^{239}N$ and are calculated by linear interpolation. The effect of resonance absorption at a specific energy on broad group removal cross sections is calculated by flux interpolation for elastic downscattering. The post processing module of the VESNA code generates constants for reactor diffusion calculation. The streaming corrections on diffusion coefficients are computed with the heterogeneous flux and Benoist's oriented first flight collision probabilities.

A large number of lattices and other experiments, mainly on reactor RB at Vinca, has been analysed and a wide range of systems could be predic-

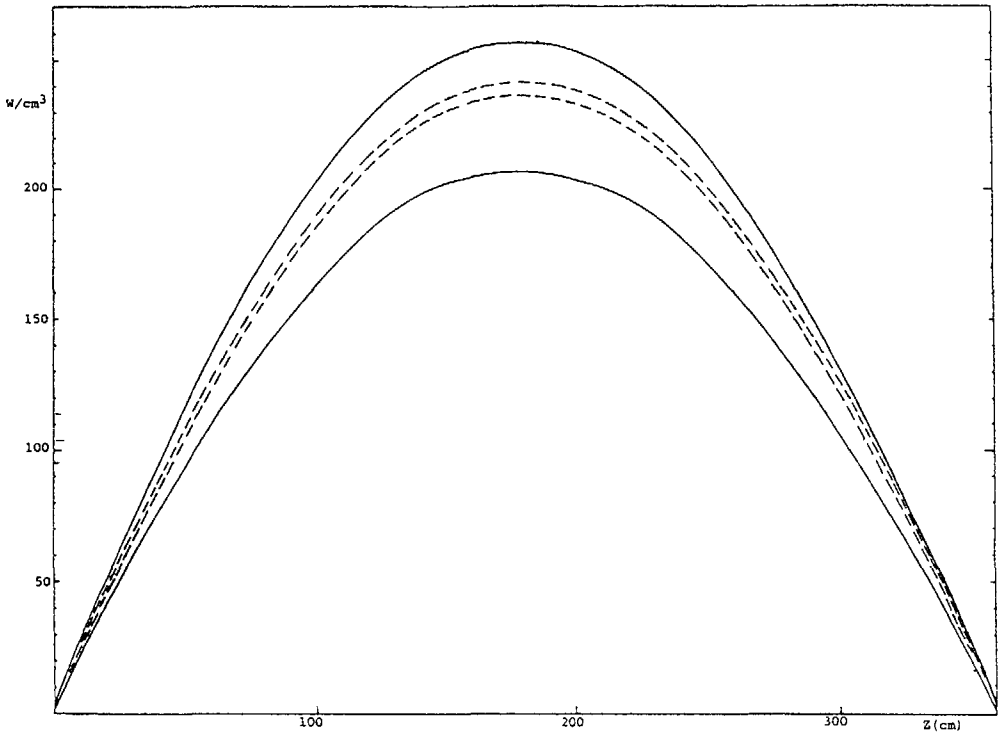

Figure 6. Hot channel axial distribution of power density, HFP core. (1) and (2) BOC and EOC of equilibrium cycle, no burnable absorbers; (3) and (4) BOC and EOC of equilibrium cycle with integral type burnable absorbers.

ted within ±1%. The development of VESNA code is now at the final stage. The goals for future work are: (1) development of methods for time dependent calculations of isotopic concentrations in an arbitrary number of media within a geometric domain, (2) use of an approximate scheme for two dimensional geometries based on the interface current approach and (3) development of a special depletion option for poison pins.

In order to improve overall reactor core calculations and criticality search, a code KONEL has been developed for solving few group 2-D diffusion equations in X-Y geometry by the finite element method. Applying Lagrangian polynomials as peacewise polunomials in a variational form of the diffusion equation, a system of linear algebraic equations is obtained. A group by group direct solution method is used. Efficiency of the direct procedure lies in assured convergence and accurate results. As the system matrix is positive definite, a Choleski decomposition is used in order to keep the computer storage under control. The numeration of nodes and finite elements is performed by line dissection, making possible the given problem to be solved as a series of smaller problems in each group. The present version of the code has no acceleration of outer iterations. At this stage of development, the output of the KONEL code are the effective multiplication factor and neutron flux, i.e. power distribution.

The KONEL code has been tested using the 2-D IAEA LWR benchmark [13], as well as some other benchmark cases. The results obtained so far agree well with the results of the finite difference code VAMPIR [8] and the finite element code FINELM [14] and justify further development and improvement of the code. As can be seen from Table VII, for the IAEA benchmark case the KONEL code provided more accurate values of the effective neutron multiplication factor than the finite difference code VAMPIR, while the calculation time was more than three times shorter.

A code called SNOOPY [9] has been developed for in core fuel management optimization by a simple and efficient linear programming method. The optimization model is based on coarse zonal discretization of a reactor core. Fuel assamblies are grouped according to their burnup. Control variable is the number of fuel elements in a certain group and in a specified zone. The objective function is the total number of fresh fuel elements to be loaded, while the constraints are the EOC reactivity and power peaking factors. The first constraint is linearized by the first order perturbation theory and the power peaking constraints are represented by empirical linear relations. The suboptimal solution is found after few linear programming iterations. The SNOOPY code has been tested by solving the in-core fuel management problems of the research reactor RA at Boris Kidric Institute.

Recently, for low-leakage PWR's a new method for optimal loading pattern search has been proposed [15]. Here optimization objective is maximal EOC reactivity, which is subject to power peaking constraints during the fuel burnup. Nonlinear problem is linearized by the use of variational formulation of depletion perturbation theory. Burnuble absorbers depletion is treated explicitely as a state equation in the development of sensitivity coefficients. Each fuel element is characterized by burnuble absorber concentration (if fresh) and burnup (if reused). Fuel shuffling is described by Boolean variables, while continuous variables are used for concentrations of burnuble absorbers. Optimization problem is reduced to a special case of quadratic mixed integer problem, since interaction of fuel shuffling and absorber optimization is also taken into account. Presently, the feasibility of the model is being tested for 1-D geometry. A prerequisite for its broader use is the availability of a very fast code for solving diffusion equations.

Table VII - Calculation of the effective neutron multiplication factor for a 2-D IAEA benchmark [13]. Comparison of programs VAMPIR [8], FINELM [14] and KOMEL [6]

	k_{eff}	Relative error (%)	Number of iterations	Computing time
Program VAMPIR				
mesh distance 5 cm	1.0266895	$28\ 10^{-2}$	205	3min 52.61sec
mesh distance 2.5cm	1.0257893	$37\ 10^{-2}$	178	18min 12.17sec
Program FINELM				
order of approx. 1	1.0451313	1.5	-	-
order of approx. 2	1.0307349	$11\ 10^{-2}$	-	-
order of approx. 3	1.0297120	$12\ 10^{-3}$	-	-
order of approx. 4	1.0295984	$8\ 10^{-4}$	-	-
Program KOMEL				
order of approx. 1	1.0305338	$9\ 10^{-2}$	80	24.19sec
order of approx. 2	1.0293713	$2\ 10^{-2}$	70	56.8 sec
order of approx. 3	1.0293779	$2\ 10^{-2}$	95	2min 44.65sec
order of approx. 4	1.0293217	$2\ 10^{-2}$	95	5min 19.57sec

5. CONCLUSION

At the Nuclear Engineering Department of the Boris Kidric Institute of Nuclear Sciences a computer code package has been assembled for in-core fuel management and related core parameter calculations. This code package has been extensively used to study advanced fuel utilization schemes in different types of power reactors, as well as for solving in-core fuel management problems of the own reasearch reactors.

Different improvements in fuel design and fuel management, introduced in order to get prolonged fuel burnup and better fuel cycle economy, require more sophisticated calculational models and codes. Particularly desirable are faster codes which make possible more detailed and numerous calculations in the frame of different optimization procedures. Efforts are thus being made at NET IBK to improve reactor cell and fuel assembly parameters calculations, to derive advanced methods for solving diffusion equations, as well as to develop optimization procedures for optimal fuel loading pattern search.

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