FUEL DEPLETION CALCULATION IN MTR-LEU NUR REACTOR

by

Foudil ZEGGAR[1], Amrane HALILOU[2], Brahim MEFTAH[2], Mohamed Y. MOKEDDEM[3], Fathi LETAIM[4], and Anis BOUSBIA-SALAH[5]

Received on February 17, 2008; accepted on May 19, 2008

In this article, we present the results of a few energy groups calculations for the NUR reactor fuel depletion analysis up to 45 000 MWd/tU taken as the maximum fuel burn up. The WIMSD-4 cell code has been employed as a calculation tool. In this study, we are interested in actinides such as the uranium and plutonium isotopes, as well as fission products Xe-135, Sm-149, Sm-151, Eu-155, and Gd-157.
Calculation results regarding the five energy groups are in a good agreement with those obtained with only two energy groups which can, therefore, be used in all subsequent calculations. Calculation results presented in this article can be used as a microscopic data base for estimating the amount of radioactive sources randomly dispersed in the environment. They can also be used to monitor the fuel assemblies inventory at the core level.

Key words: MTR-LEU fuel, burn up, NUR reactor, WIMSD-4

INTRODUCTION

The isotopic composition inventory of nuclear fuel is in a state of continuous evolution during the operating lifetime of a nuclear reactor. It is the result of fuel depletion or burn up due to nuclear reactions. Fuel irradiation induces long-term changes in its isotopic composition which should be determined by means of calculation.

The main parameters accompanying this phenomenon are the distribution of power densities, the variation of the flux, the power picking factors, and reactivity. All these variations strongly influence the operating life, stability, and control of a reactor.

To compensate for the loss of reactivity due to fuel depletion, a high-enough initial excess of reactivity should be provided, ensuring that the initial total mass of uranium is greater than the critical mass.

Among the phenomena that negatively influence total reactivity, we can cite the effect of temperature, poisoning by Xe-135 and Sm-149 and fuel depletion [1].

The conversion factor, which is a positive effect, is too small in thermal research reactors to compensate for the negative effects.

The quantity and activity of each fission product during the reactor's operation can be used to evaluate: the radioactive sources randomly dispersed in the environment, the amount of fuel element activity after its discharge from the core, the residual heat in a fuel element, and the parasitic capture of neutrons by fission products accumulated during the reactor's operation. This analysis has focused on the two main fission products, Xe-135 and Sm-149.

The change in the isotopic composition with burn up causes changes in the macroscopic cross- sec-

Scientific paper
UDC: 539.125.52:621.039.516/.517
BIBLID: 1451-3994, *23* (2008), 1, pp. 11-18
DOI: 10.2298/NTRP0801011Z

Authors' addresses:
[1] Division sûreté nucléaire et radioprotection,
Centre de recherche nucléaire de Draria (CRND)
BP 43, Sebala Alger, Algeria
[2] Division Réacteur,
Centre de recherche nucléaire de Draria (CRND)
BP 43, Sebala Alger, Algeria
[3] Division physique et applications nucléaires,
Centre de recherche nucléaire de Draria (CRND)
BP 43, Sebala Alger, Algeria
[4] Faculté de Physique, Université Haouri Boumediene (USTHB)
BP 31, Bab Ezzouar, Alger, Algeria
[5] Dipartimento di Ingegneria Meccanica,
Nucleari e della Produzione,
Facoltà di Ingegneria,
Università di Pisa,
2, Via Diotisalvi, 56126 Pisa, Italy

E-mail address of corresponding author:
zeggar_fod@yahoo.fr (F. Zeggar)

tions which should be evaluated by a transport code in each burn up step.

For that purpose, the cell code WIMSD-4 [2, 3] has systematically been used in our study. Nuclide chain equations are solved in the homogenized equivalent cell by using the diffusion theory.

The main objective of this work is to present and discuss the fuel lattice depletion analysis approach used at NUR.

NUR REACTOR CHARACTERISTICS

NUR is a 1 MW nuclear research reactor moderated and cooled with light water [4]. It is an open swimming pool reactor using U_3O_8 MTR type fuel enriched to 19.7%. Light water and blocks of graphite are used as reflectors. The fuel elements are fixed on an oblong grid which permits a variable configuration arrangement.

A schematic diagram giving the cross-sectional view of the reactor core is shown in fig.1. Figures 2 and 3 show schematic views of the standard and control fuel elements, respectively. Table 1 gives the NUR reactor main data.

The reactor went critical on March 23rd, 1989.

REVIEW OF LATTICE BURN UP AND DEPLETION CALCULATION IN WIMSD-4

In this section, a brief review of WIMSD-4 calculation methods for local burn up calculations is given. The main reference from which the data are taken is that of Kulikowska [5].

WIMSD-4 performs burn up calculations in module chains 12 and 16. A POWERC input card initiates the process. The general scheme of burn up calculations in WIMSD-4 is given in fig. 4.

Lattice calculations are executed for a steady-state reactor and the time variation is not explicitly taken into account. Thus, the macroscopic cross-sections which are the coefficients of the transport equation are considered to be constant within each time interval.

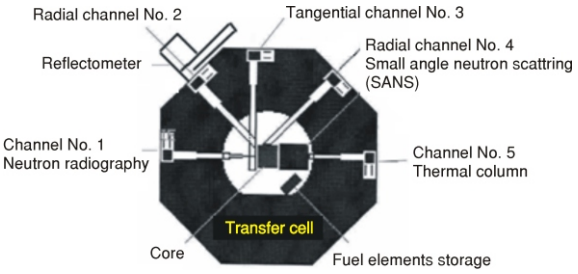

Figure 1. Cross-sectional view through NUR reactor core

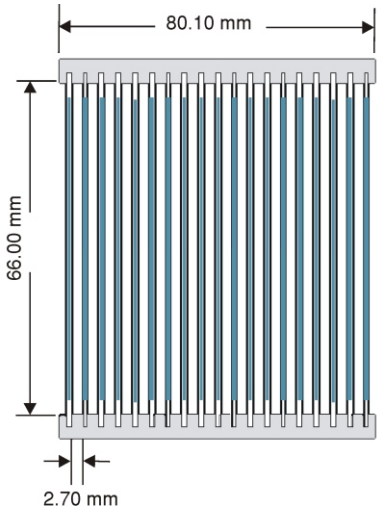

Figure 2. Standard fuel element

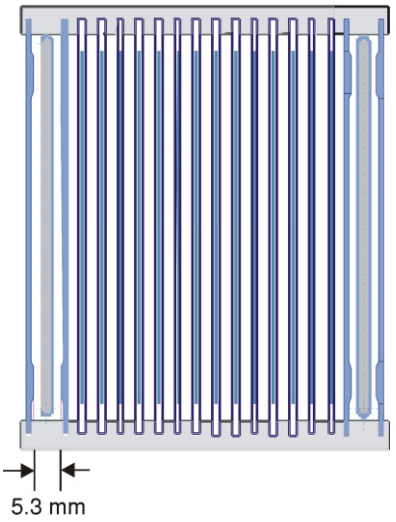

Figure 3. Control fuel element

WIMSD-4 is set out to perform burn up analysis on a lattice unit cell.

Burn up calculations constitute the last step of a WIMS run. They are performed in chains 12 and 16 and are executed if a POWERC control card (which defines a mean power density RQ, a time step between criticality calculation RTAU, and the number INDNB of such time steps between lattice calculations) is included in the main data group of input cards.

The various computational steps performed are:

(1) Preparation of tables of effective absorption and fission microscopic cross-sections for the nuclides of each material undergoing burn up on the basis of:
   (a) code library microscopic data, and
   (b) resonance data and fluxes resulting from the spectrum calculation step;

Table 1. NUR fuel elements main data

Standard fuel element (SFE)	
Fuel type	U_3O_8-Al
Enrichment	20%
Uranium density in the meat	2.96 g/cm^3
Average U plate content	76.5 g
Void fraction in the meat	10%
U_3O_8 density	8.1 g/cm^3
Uranium weight fraction in the U_3O_8	0.8477
Al density	2.7 g/cm^3
Number of fuel plates	19
Plate thickness	0.150 cm
Meat thickness	0.070 cm
Clad thickness	0.040 cm
Water channel thickness	0.270 cm
Meat width	6.0 cm
Meat height (or active height)	61.5 cm
Frame thickness	0.45 cm
Inner distance between frames	6.7 cm
Frame length	8.01 cm
Structural material (frames and clads)	Al-6061
Control fuel element (CFE)	
Fuel plate	same as SFE
Number of fuel plates	14
Water channel thickness	same as SFE
Frame thickness	0.45 cm
Inner distance between frames	6.7 cm
Frame length	8.01 cm
Absorber material	Ag-In-Cd (80%, 15%, 5%)
Clad material for absorber	ANSI 316
Material gap within the absorber	He
Guide plates material	Al-6061
Grid dimensions	7.7×8.1 cm

(2) Preparation of macroscopic cross-sections with absorption and fission cross-sections corrected for number densities modification due to burn up;
(3) Solution of the few group criticality equation, with the appropriate control option for the homogenized cell;
(4) Flux renormalization to the required power level and calculation of reaction rates (absorption, capture, and fission) for each burnable nuclide in each burnable material;
(5) Numerical integration of the following nuclides concentration equations

$$\frac{dN_i}{dt} \quad \lambda_i N_i(t) \quad A_i N_i(t) \quad \sum_k \delta[i, j_1(k)]\alpha_{ki} C_k N_k(t) \quad \sum_k \delta[i, j_2(k)]\beta_{ki}\lambda_k N_k(t) \quad \sum_k Y_{ki} F_k N_k(t) \tag{1}$$

where N_i is the density of the nuclide i, λ_i – the decay constant of the nuclide i, A_i – the microscopic absorption reaction rate for the nuclide i, F_k – the microscopic fission reaction rate for the nuclide k, C_k – the microscopic capture reaction rate for the nuclide k,

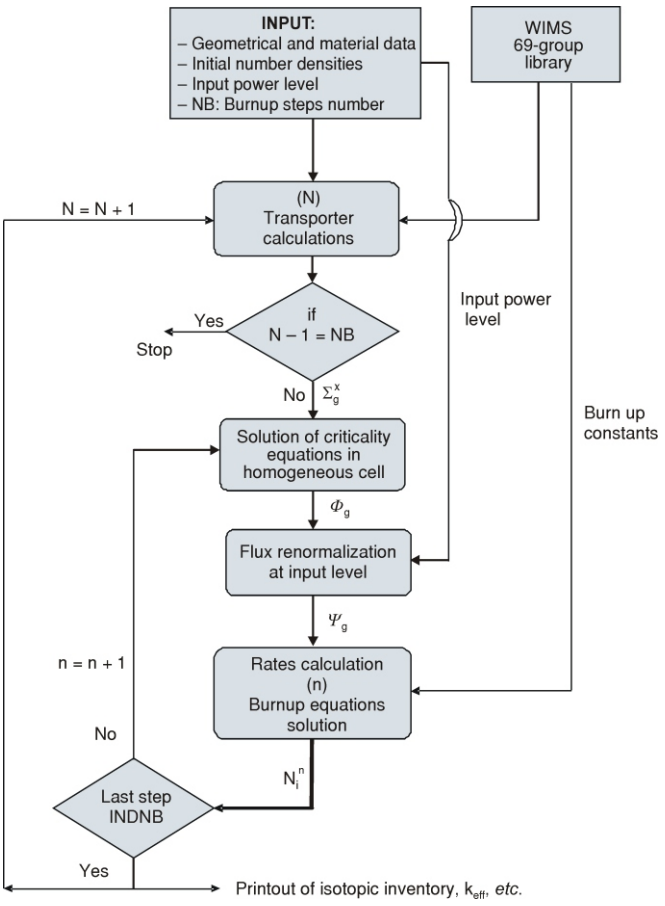

Figure 4. General scheme of burn up calculation in WIMSD-4

Y_{ki} – the fission yield of the nuclide i for a fission of the nuclide k, α_{ki}, β_{ki} – the fractions of the nuclide i issued from capture or radioactive decay of the nuclide k, respectively, $j_1(k)$, $j_2(k)$ – the products identifiers of the nuclide k by capture or radioactive decay, respectively, and $\delta(i, j)$ – the Kronecker symbol indicating that the contribution occurs if i = j.

CALCULATION CONDITIONS, RESULTS AND DISCUSSION

Calculation conditions

The NUR core includes two types of fuel elements: standard fuel elements (SFE) and control fuel elements (CFE). The SFE is made of 19 fuel plates, while the CFE includes 14 fuel plates. The calculations were made in 2 and 5 energy groups based on the 69-WIMS library [6]. Table 2 shows the detail of the energy structure adopted for the libraries. The list of nuclides employed in the present burn up calculation is shown in tab. 3 for actinides and fission products, respectively. WIMS executes burn up calculations in CHAINS 12 and 16 when a POWERC input card is given. The calculations were performed for both a cold state core (T = 20.0 °C for all materials) and a hot state core (T = 63 °C for fuel material, T = 52.6 °C for Al-clad material, and T = 42.2 °C for the H_2O moderator and all others materials).

A boron equivalent quantity of 8 ppm/g is added to the U_3O_8-Al meat and 48 ppm/g to the structural aluminum in order to account for the presence of impurities in these materials [7].

Table 2. Two and five energy groups structure

	Group	Energy (eV)	
		Upper energy boundary	Lower energy boundary
2G	1	1.000E+7*	0.625
	2	0.625	0.0
5G	1	1.000E+7	0.821E+6
	2	0.821E+6	5530.0
	3	5530.0	0.625
	4	0.625	0.080
	5	0.080	0.0

* E+7 read as 10^7

Table 3. Nuclides considered in the present burn up calculation

Actinides	Fission products
U-235	Cd-113
U-236	Xe-135
U-238	Sm-149
Pu-239	Sm-151
Pu-240	Eu-155
Pu-241	Gd-157

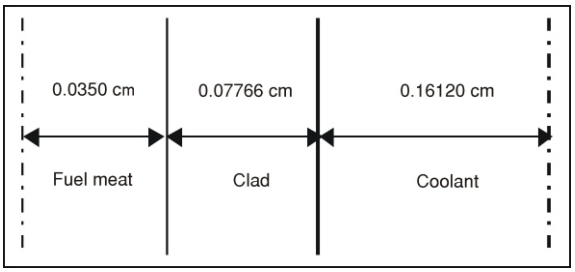

Figure 5. Calculation cell of NUR reactor core

The equivalent cell model adopted in the calculations is a three regions model, as recommended in ref. [8]. The geometrical representation of such a cell is given in fig. 5.

It is seen that region (2) includes the Al-clad and the structural aluminum of the fuel assembly. Region (3) includes H_2O of the inter-plates and that of surrounding channels.

Results and discussion

The results of WIMSD-4 fuel lattice depletion analysis for the three regions lattice equivalent cell of the NUR standard fuel assembly in two and five energy groups are presented in figs. 6-8 and tab. 4. All the results are obtained by assuming the reactor is operating at a constant power of 1 MW.

The nuclear fuel depletes with burn up and changes occur in its isotopic composition. This has an influence on the neutron spectrum. The effects on the NUR fuel can be seen in fig. 6, where the isotopic depletion of U-235 and U-238, as well as the build-up of U-236, Pu-239, Pu-240, and Pu-241 are observed through the variation of their atom number densities with burn up. These results show a decrease of densities for U-235 and U-238 as the fuel depletes. We also notice that after a year of reactor operation at a 1 MW constant power, only 10% of the initial fissile material is consumed. In the opposite case, an increase is observed for other actinides. The examination of the curves in fig. 6 shows that the various plutonium isotopes appear one after another, because Pu-240 is produced from Pu-239 by neutronic capture, Pu-241 from Pu-240, and so on.

Results in fig. 7 show the evolution of fission products that have an influence on reactivity, notably the Xe-135. Its concentration increases rapidly and saturates just after a few days of reactor operation, then it decreases slowly with burn up. The effect of burn up on reactivity is given in fig. 8 and tab. 5. Apart from the beginning of the curve which corresponds to the apparition of Xe-135, the decrease of the cell reactivity conforms to the reactor operating time or burn up. For the core conditions: cold, fresh, and without Xe-135, the re-

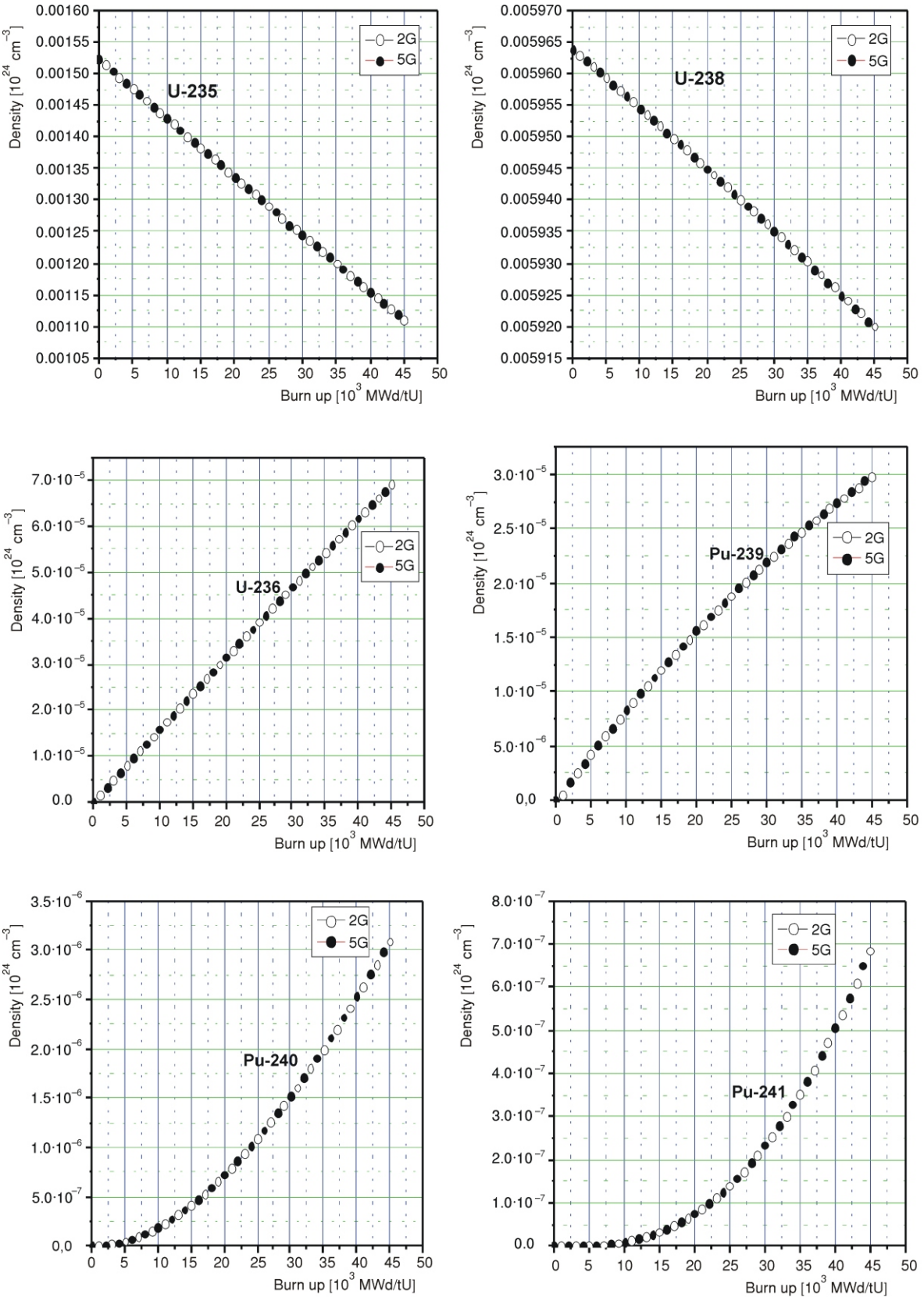

Figure 6. Actinide inventory: atom number density *vs*. fuel burn up

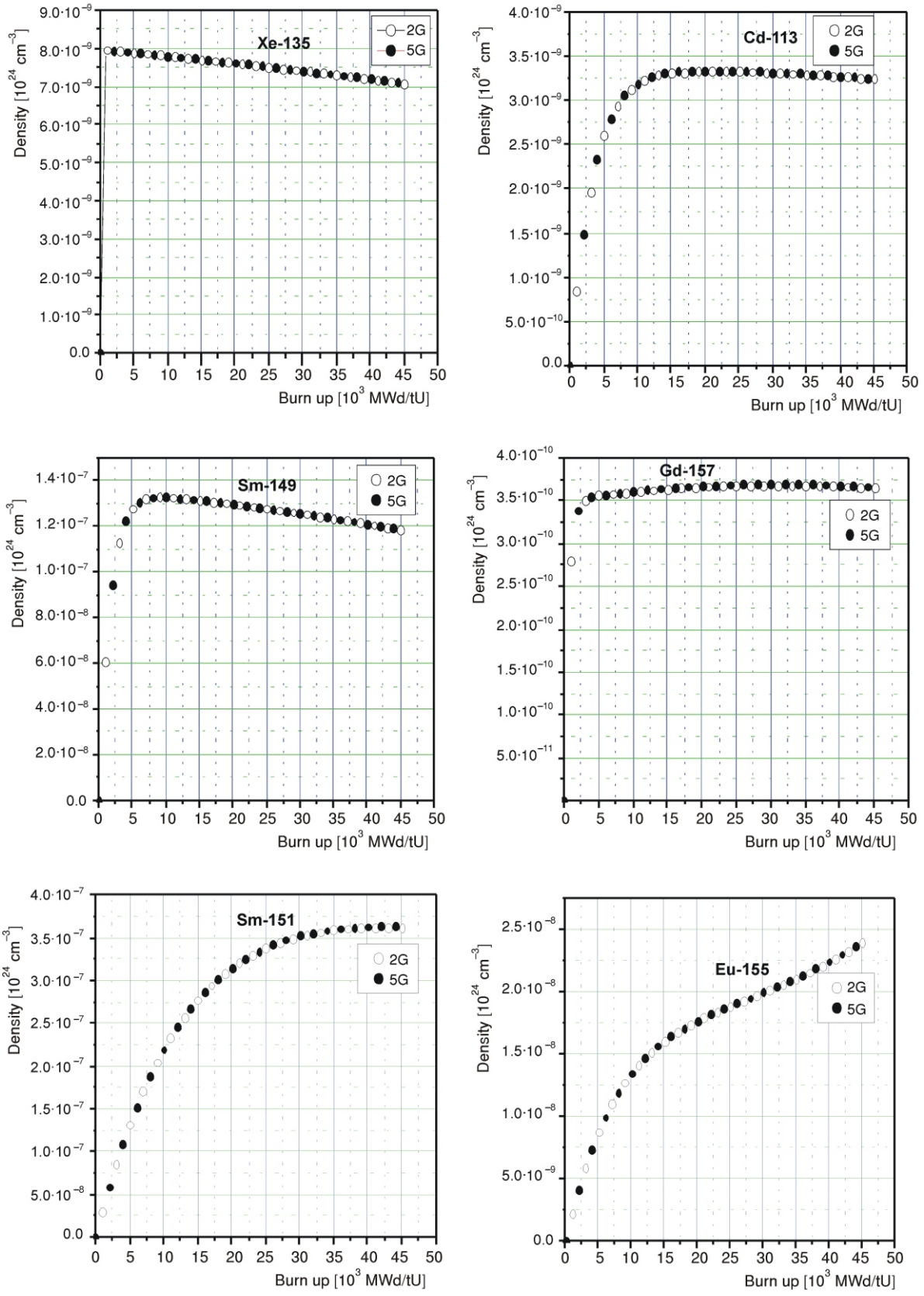

Figure 7. Fission products inventory: atom number density *vs.* fuel burn up

Table 4. Variation of isotopic densities with burn up for major fissile, fertile, and poison isotopes

Burn up [MWd/tU]	Isotope	Density [10^{24} cm^{-3}]		Error [%]
		2 groups	5 groups	
1000	U-235	0.0015148	0.0015148	0.000
	U-238	0.0059631	0.0059663	0.000
	Pu-239	8.6978E-7	8.7389E-7	0.470
	Pu-241	1.0680E-11	1.0759E-11	0.730
	Xe-135	7.9639E-9	7.9639E-9	0.000
	Sm-149	6.0956E-8	6.0973E-8	0.027
3000	U-235	0.0014959	0.0014959	0.000
	U-238	0.0059613	0.0059612	0.000
	Pu-239	2.5826E-6	2.595E-6	0.470
	Pu-241	2.8614E-10	2.8825E-10	0.730
	Xe-135	7.9275E-9	7.9321E-9	0.057
	Sm-149	1.1271E-7	1.1276E-7	0.035
5000	U-235	0.0014771	0.0014771	0.000
	U-238	0.0059594	0.0059594	0.000
	Pu-239	4.2769E-6	4.2769E-6	0.000
	Pu-241	1.3219E-9	1.3219E-9	0.000
	Xe-135	7.8982E-9	7.8982E-9	0.000
	Sm-149	1.2789E-7	1.2789E-7	0.000
7000	U-235	0.0014583	0.0014583	0.000
	U-238	0.0059575	0.0059575	0.000
	Pu-239	5.918E-6	5.9184E-6	0.000
	Pu-241	3.5822E-9	3.5822E-9	0.000
	Xe-135	7.863E-9	7.8634E-9	0.000
	Sm-149	1.3201E-7	1.3201E-7	0.000
9000	U-235	0.0014396	0.00144396	0.000
	U-238	0.0059557	0.0059556	0.000
	Pu-239	7.4846E-6	7.5194E-6	0.460
	Pu-241	7.4529E-9	7.5073E-9	0.720
	Xe-135	7.8231E-9	7.8279E-9	0.060
	Sm-149	1.3277E-7	1.3283E-7	0.045
11000	U-235	0.0014209	0.00142	0.000
	U-238	0.0059537	0.0059537	0.000
	Pu-239	9.0386E-6	9.0804E-6	0.460
	Pu-241	1.3406E-8	1.3504E-8	0.720
	Xe-135	7.787E-9	7.7918E-9	0.350
	Sm-149	1.326E-7	1.3267E-7	0.051

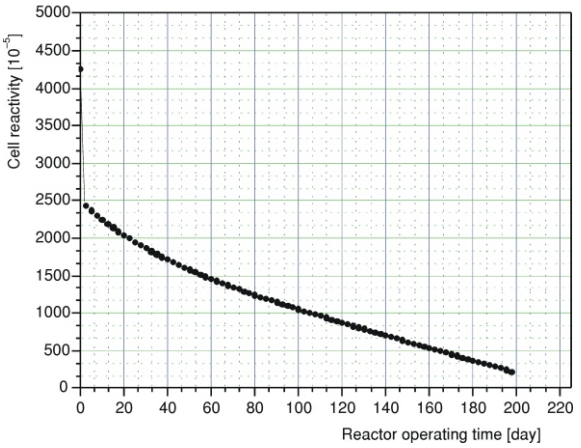

Figure 8. NUR reactivity variation at 1 MW constant power

activity excess is seen to be 4273 10^{-5} pcm. This value is high enough to compensate for the decrease in reactivity with burn up for a cycle greater than 100 days. The lowest value will actually depend on both experimental needs and the operability of the reactor.

Table 5. Variation of reactivity with NUR reactor operating time

NUR reator operating time [day]	Burn up [MWd/tU]	Reactivity [10^{-5}]
0	0	4273
0.25	11.75	2503
0.5	23.5	2495
0.75	35.25	2488
1	47	2480

CONCLUSION

The isotopic inventory of some actinides and fission products in the NUR nuclear research reactor have been estimated for a fuel cell of the MTR type enriched to 19.7%, using the cell code WIMSD-4 based on the WIMS-69 energy group library.

The fuel depletes with burn up, leading to a significant change in its composition and reactivity. The analysis allowed us to give an estimation of the isotopic concentrations of actinides and some fission

products present in an average fuel assembly in function of the reactor operating time or burn up. It also gave an estimation of the decrease of the reactivity as a function of time. The initial reactivity excess of 4273 10^{-5} drops to a value of 500 10^{-5} after about 150 days of reactor operation. This gives us an approximation of the reactor's fuel cycle period.

REFERENCES

[1] Duderstadt, J. J., Hamilton, L. J., Nuclear Reactor Analysis, John Wiley and Sons, New York, USA, 1976

[2] Askew, J. R., Fayers, F. J., Kemshell, P. B., A General Description of the Lattice Code WIMS, UKAEA, 1967

[3] Halsall, M. J., Summary of WIMSD-4 Input Option, AEEW-M 1327, 1980

[4] ***, INVAP SE, Réacteur nucléaire pour formation – Rapport final de sûreté, Haut Commissariat à la Recherche, Algiers, March, 1989

[5] Kulikowska, T., Reactor Lattice Transport Calculation, Workshop on Nuclear Data and Nuclear Reactors, Physics, Design and Safety, Trieste, Italy, March-April, 2000

[6] Taubman, C. J., The WIMS 69-Group Library Tape 166259, AEEW-M1324, 1975

[7] Madariaga, M., Rubio, R., Calculo neutronico del nucleo de arranque del RAE, Informe subgerencia de sistemas, INVAP SE, San Carlos de Bariloche, Argentina, 1988

[8] Meftah, B., Zidi, T., Bousbia-Salah, A., Neutron Flux Optimization in Irradiation Channels at NUR Research Reactor, *Annals of Nuclear Energy, 33* (2006), 14-15, pp. 1164-1175

**Фудил ЗЕГАР, Амран ХАЛИЛУ, Брахим МЕФТАХ,
Мохамед Ј. МОКЕДЕМ, Фати ЛЕТАИМ, Анис БУСБИА-САЛАХ**

ПРОРАЧУН УТРОШКА ГОРИВА У МТР-ЛЕУ НУР РЕАКТОРУ

У раду се приказују резултати вишегрупног енергетског прорачуна спроведеног ради анализе утрошка горива НУР реактора до максималног изгарања од 45 000 MWd/tU. Као средство прорачуна коришћен је ћелијски програм WIMSD-4. Интерес у овом изучавању је за актиниде, као што су уранијумови и плутонијумови изотопи и фисиони продукти Xe-135, Sm-149, Sm-151, Eu-155 и Gd-157. Резултати петогрупног енергетског прорачуна у доброј су сагласности са онима добијеним двогрупно, те је двогрупни прорачун коришћен у свим каснијим прорачунима. Приказани резултати прорачуна могу се користити као микроскопска база података за процену количине радиоактивних извора насумице расутих у животну средину. Могу се такође употребити за надгледање стања горивних склопова у језгру.