NUKLEONIKA 2005;50(3):113–120

ORIGINAL PAPER

Evaluation of the activity of irradiated graphite in the Ignalina Nuclear Power Plant RBMK-1500 reactor

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Abstract Ignalina NPP Unit 1 with the RBMK-1500 type reactor enters the preparatory stage for its decommissioning. A big volume of graphite has to be dismantled and managed. This paper evaluates the composition of radionuclides in irradiated graphite on the basis of the calculation scheme with computer codes MCNPX and CINDER'90. Full scale calculations take into account the whole energy spectrum of neutrons, the spatial neutron flux in different reactor structures and the impurities of RBMK-1500 graphite measured by two independent methods. Principal contributors to the total activity as well as other radionuclides important from the radiological point of view are identified. The uncertainties of such calculations are discussed.

Key words modeling • activation analysis • radioactive waste • graphite

Introduction

Determination of radiological characteristics of graphite structures after the operational period of nuclear reactors which use graphite as a moderator or reflector is a highly important issue from the point of view of their decommissioning. Good description of the inventory of radionuclides directly implies the choice of dismantling and residue management technologies, which allows the minimization of volumes of radioactive waste and costs of its management.

The composition of radionuclides in reactor graphite structures (with exception of the accidental contamination due to fuel element rupture) depends on impurities and irradiation variables, namely the spatial flux of neutrons, activation and cooling time. The spatial flux can be obtained from deterministic solution of complex integro-differential neutron transport equations or can be obtained by Monte Carlo approach. The irradiation of impurities in the neutron flux results in the chains of nuclear reactions, which may be described by methods used for Markov processes. Here the computer codes also found successful application. Preliminary numerical evaluation of the radionuclide composition in the RBMK-type reactor graphite may be found in [14]. This work is based on the provisionally chosen thermal neutron flux and the impurities data taken from the scientific literature. The amount of impurities rates, as reported, significantly influences the calculation results. In addition, special attention should be paid to the neutron flux variations because of the use of different fuel elements and their burnup as investigated in [12]. Finally, the activation estimations of some individual radionuclides depend on the uncertainty due to different neutron effective cross-section

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Received: 3 January 2005

libraries used by the computer codes [9]. In [8], the induced radioactivity in the Chernobyl Nuclear Power Plant with RBMK-1000 type reactors was evaluated using a similar approach.

This paper presents the results of estimation of spatial graphite activation of the Ignalina Nuclear Power Plant RBMK-1500 type reactor. Our study takes into account neutron flux variations occurring in respective graphite constructions due to the power load variations, fuel burnup and insertion of control rods. The data on impurities of graphite used in the Ignalina NPP reactors, investigated by two independent experimental techniques, have been used for the activation calculations. Finally, the full scale Monte Carlo calculations take into account the whole energy spectrum of neutrons.

General data on Ignalina NPP reactors and methodology

There are two RBMK-1500 type reactors in Ignalina site situated in the north-eastern part of Lithuania. RBMK-1500 is a channel type, boiling light water reactor, which uses graphite as a moderator-reflector and has a core of cylindrical shape. The core is surrounded by circular (here referred to as the side reflector), top and bottom graphite reflectors. The total mass of graphite per reactor unit is more than 1800 tons. Unit 1 started its operation in 1984 and Unit 2 – in 1987. The reactor is refueled during operation and each unit has an outage of approximately 2 months annually. Table 1 presents general data of the Ignalina NPP reactors [1].

The overall calculation scheme used for this work is presented in Fig. 1. The activation calculations are carried out in 2 phases: the first phase is the calculation of the spatial neutron flux distributions and energy spectra using MCNPX [4] Monte Carlo code; the second phase is the calculation of activation, radioactive decay and cooling. The MCNPX input data comprise: nuclear data, materials specifications and detailed geometry description. The continuous neutron energy spectrum obtained with MCNPX was divided into 63 energy groups and normalized to a nominal power of the reactor (4250 MW). The spatial neutron flux is further used as the input data for the calculations in the second (activation) phase. To obtain the irradiation scenario which is close to reality, the history of the power load of Unit 1 during 21 years was used by the transmutation code CINDER'90 [17] in time steps averaged over one year. The code also needs the chemical composition and impurities of materials. Two independent studies for the determination of the RBMK-1500 graphite impurities have been carried out in the Saclay Research Centre of French Atomic Energy Commission (CEA): gamma spectroscopy based on activation by neutrons and glow discharge mass spectroscopy (GDMS).

Table 1. General data of the RBMK-1500 reactor

Nominal thermal power	MW	4250
Nominal electrical power	MW	1300
Core height	m	7
Core diameter	m	11.8
Lattice pitch	m	0.25×0.25
Top/bottom reflectors thickness	m	0.5
Side reflector thickness	m	0.88
Number of fuel channels		1661
Fuel/enrichment of ^{235}U, %/burnup, MWd/kg		uranium dioxide; 2.0*; 21.6**
Maximum acceptable graphite stack temperature	°C	760
Maximum acceptable fuel channel temperature	°C	650

* At present the fuel is changed to 2.4% enrichment fuel with Er as a burnable poison.

** At fuel enrichment for ^{235}U of 2.0%.

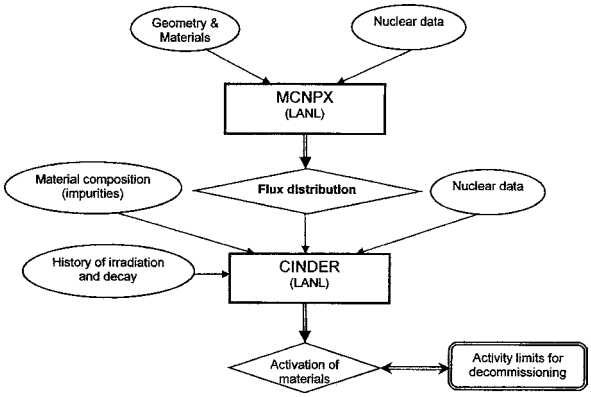

Fig. 1. A schematic procedure for activation calculations.

The CINDER'90 code uses 63 neutron energy groups and has its own nuclear data library composed mainly of ENDF, JEF and JENDL data libraries. The output of the code is: the isotopic composition and radionuclide activity in the irradiated material. The calculated activities are compared with the established activity clearance levels for the materials during decommissioning [11].

The theoretical calculations of the spatial neutron flux distribution using the MCNPX code have been already validated for the RUS research reactor at Strasbourg [10]. They are in good agreement with the experimental flux measurements. Activation calculations for the same reactor using the above scheme were also in excellent agreement with the experimental measurements both for the graphite and different types of concrete.

Results

Analyses of the impurities of the RBMK-1500 graphite

The specimen of the fresh RBMK-1500 graphite taken from the fuel channel sleeve has been analyzed. Table 2

Table 2. The results of impurity of the RBMK graphite specimen (also see text for details)

Impurity	Concentration ppm	Impurity	Concentration ppm	Impurity	Concentration ppm
Li	0.004–0.05 [7, 16]	Ni	0.39	La	0.15
Be	0.02 [3]	Cu	*0.1*	Ce	0.269
B	*0.05*	Zn	0.02	Pr	0.08
N	0.5–70 [6, 7]	Ga	0.01	Nd	0.11
O	40–197.5 [9]	Ge	9.0	Sm	0.0213
Na	4.64; *5.0*	As	0.011	Eu	0.0026
Mg	7.0; *0.5*	Se	0.003	Tb	0.0027
Al	9.2; *1.0*	Br	0.025	Dy	0.0032
Si	*1.0*	Rb	0.008	Ho	0.0094
P	*0.5*	Sr	0.96	Er	0.0053
S	5–52 [5]	Zr	1.0	Tm	0.0056
Cl	7.6	Mo	0.17	Yb	0.014
Ar	0.14	Ru	0.07	Lu	0.0015
K	1.9; *1.5*	Ag	0.003	Hf	0.0058
Ca	51.9; *2.0*	Cd	0.015	Ta	0.0019
Sc	0.05	In	0.003	W	0.047
Ti	17.4	Sn	0.15	Re	0.0019
V	17.4	Sb	0.004	Au	0.00022
Cr	0.6; *0.3*	Te	0.014	Hg	0.00062
Mn	0.58; *0.2*	I	0.04	Th	0.0079
Fe	18.7; *1.0*	Cs	0.0016	U	0.016
Co	0.019	Ba	2.01		

presents the results of the analyses. The graphite specimens were irradiated by the thermal and fast neutron flux in the CEA research reactors ORPHEE and OSIRIS ($\Phi_{th} = (1.2–2.5) \times 10^{13}$ n ($cm^{-2} s^{-1}$) for ORPHEE and $\Phi_{fast} = 2 \times 10^{13}$ n ($cm^{-2} s^{-1}$) for OSIRIS). After irradiation, the specimens were processed and analyzed by gamma spectrometry. This method enables to identify the majority of chemical elements, starting with $Z > 11$, with precision of the order of $10^{-13} – 10^{-9}$ g g^{-1} [13]. However, some important impurity elements such as Li, N, S, Nb, Pb may not be quantified by the gamma spectrometry method.

The given in italics impurity concentration data in Table 2 represent the results of GDMS analysis. This method is based on sputtering the atoms to plasma of cathode which is made from the material to be analyzed. The atoms of the material are subject to ionization by electrons ejected to argon gas plasma due to a potential difference of 500–1500 V or Penning ionization due to ionization by argon atoms. The ionized atoms of the material are then analyzed with a mass spectrometer. Although the method is rather sensitive (the limit of detection is below ($(0.05–2) \times 10^{-7}$ g) for the elements given in Table 2), some elements as Li, N, O, Cl, F may not be quantified because of the absence of Penning ionization in argon.

Both methods enabled to obtain the data on impurities present in graphite. However, for more complete impurity composition, some important elements as Li, N and others, which could not be quantified by the above methods, have been taken from scientific literature, mainly on the typical RBMK graphite composition. For these additional data the source is indicated in brackets (see Table 2).

Calculation of the neutron fluxes in graphite constructions of the RBMK-1500 reactor

MCNP results for RBMK criticality calculations, as reported in [2], have shown good agreement with experimental measurements made for the RBMK critical test facility at the Russian Research Center "Kurchatov Institute". The best results were obtained using the ENDF/B-VI nuclear data library. Therefore, our MCNPX model for the Ignalina NPP reactors uses nuclear data from this library. The model takes into account the detailed geometrical parameters of the RBMK-1500 reactor. The temperatures of graphite and water are chosen from the available list in nuclear data libraries and are, correspondingly, equal to 800 K and 600 K, which are close to the operational temperatures. The density of water in our model is 0.5 g cm^{-3} and of graphite – 1.68 g cm^{-3}. The cell of fuel assembly and

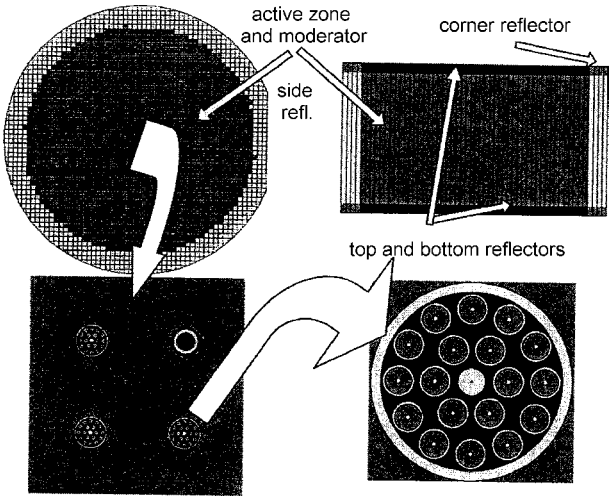

Fig. 2. The MCNPX model of the RBMK-1500 reactor with horizontal and vertical sections (top), 4 lattice elements (bottom) containing: the moderator, fuel channel sleeves, 3 fuel channels with 18 fuel rods, 1 control rod.

graphite column has been modeled as precisely as possible taking into account data given in the technical description [1]. The calculations have been carried out for the composition corresponding to fresh and irradiated fuel, the burnup of the latter being 10 MWd/kgU. In both cases the fuel with initial 2.0% ^{235}U enrichment was used. The horizontal and vertical sections of the RBMK-1500 reactor, as modeled by MCNPX, are given in Fig. 2. Five graphite constructional zones are identified: the active zone and the moderator of the reactor, the side reflector, the top and bottom reflectors, the part of reactor, where the flux of neutrons is the lowest and nearly fully thermalized (here referred to as the corner reflector), and the fuel channel graphite sleeves, which surround the fuel channel and are designed to fill the space between the channel and moderator (cylinder around the fuel channel). The neutron flux and its energy spectrum are calculated in each of the above mentioned zones. Figure 3 represents normalized distribution of neutron fluxes per unit lethargy in the RBMK-1500 reactor graphite constructions. The fastest neutron flux is observed in the fuel channel sleeves. The share of thermal neutrons is about 48% of the total flux in the sleeves as they are the closest constructional element to the fast neutrons source –

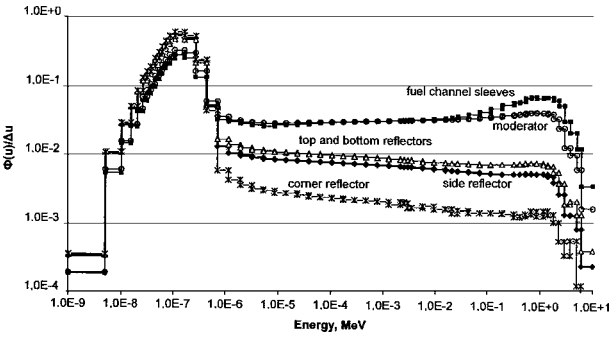

Fig. 3. The spectrum of neutron flux in different graphite constructions of the RBMK-1500 reactor.

the fuel. The thermal neutron flux share is about 56% in the moderator and more than 90% in the reflectors. To evaluate how the neutron flux and its energy distribution in moderator are sensitive to fuel burnup and the presence of control rods, three models having the same basic geometry and other parameters have been considered (Fig. 4). Model I uses fresh fuel and has no control rods. The fresh fuel is replaced with the fuel of 10 MWd/kgU burnup in model II. Model III has the same burnup as in model II, but has both fully and partially inserted control rods in the core. The maximal values of the fluxes obtained in models I and II are in the center of the reactor and the flux monotonically decreases towards the boundaries of the core (Fig. 4a). Due to the fact that the thermal power distribution in the reactor core is proportional to the fission rate, consequently, it represents the distribution of the thermal neutron flux in the reactor core. However, the total neutron flux averaged over an element of the reactor

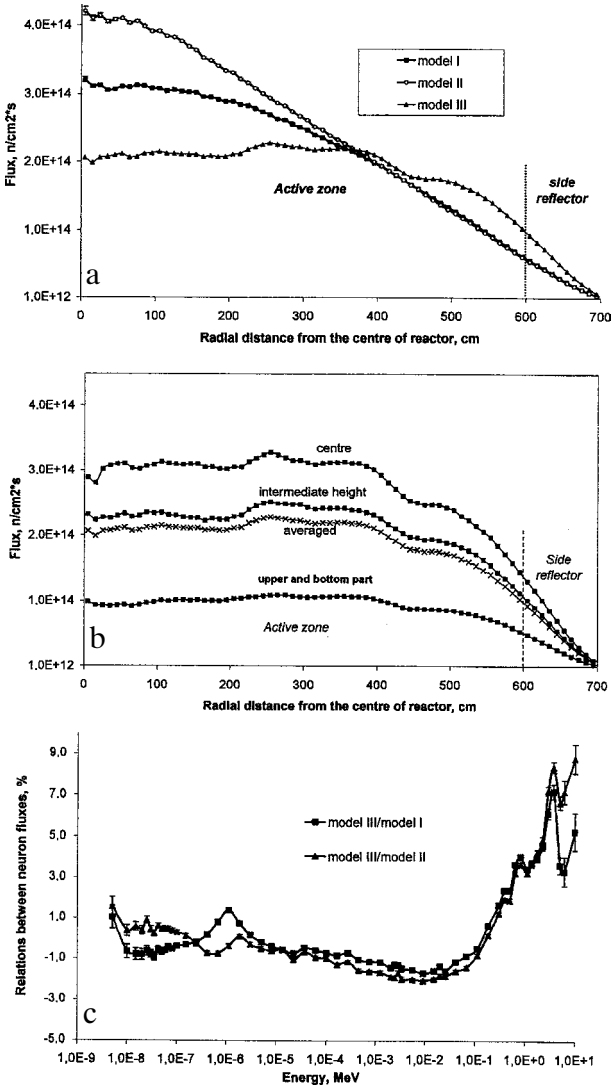

Fig. 4. a – Radial distribution of the neutron flux in the moderator and side reflector (model dependence); b – radial distribution of the neutron flux in moderator and side reflector in model III (height dependence); c – ratios between neutron fluxes for moderator in models I–III.

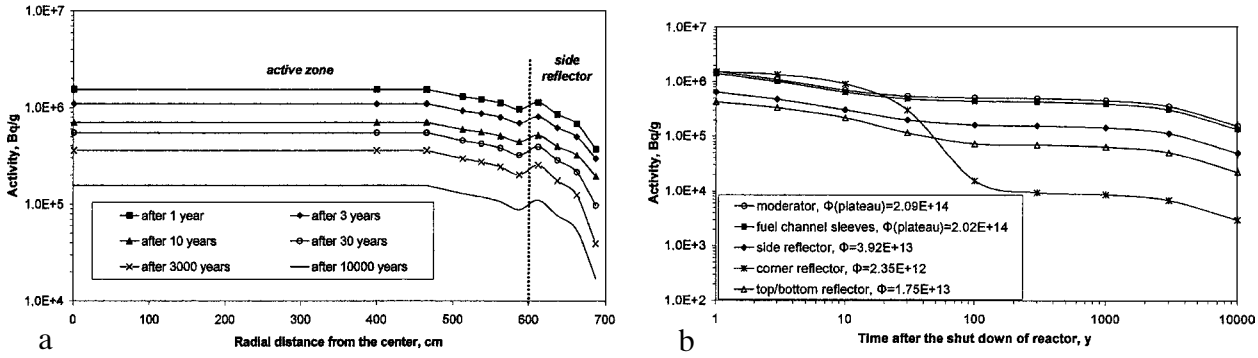

Fig. 5. Activity in the RBMK-1500 reactor as a function of cooling time: a – radial distribution of activity in moderator; b – the activity in all graphite constructions, the flux Φ is in n(cm^{-2} s^{-1}).

volume, which is large enough to contain several identical lattice elements, is also quasi-proportional to the same thermal power distribution. Therefore, the qualitative fitting of the calculated curves of axial or radial distribution of the total neutron flux in the reactor core can be obtained from the corresponding curves of the thermal power distribution. Comparing the calculated axial total neutron flux curve with the curve of the axial thermal power of the reactor from [1], we obtain that the variations are less than 30%, for the radial flux of neutrons – less than 15%. The possibility of improving the model further faces difficulties because even smaller parts than halves of the control rods including the axial fuel burnup profile should be simulated, which makes it too sophisticated and costly in terms of computer time. Figure 4c shows a comparison of the neutron energy spectra of models I and II with model III. In the models without control rods differences are less than 1% for thermal neutrons and less than 9% for fast neutrons. It allows us to conclude that deviations of the neutron flux spectrum in graphite structures because of the fuel burnup and the changing positions of control rods may not lead to a significant error in activation calculations. The fluxes calculated using model III are used in the following for the activation calculations as they represent realistic magnitude of the neutron flux in the periphery of the reactor core and in reflectors.

Other parameters, which may influence the flux calculations, are the variable density of the water-steam mixture in the fuel channel and the use of higher enriched erbium fuel (Table 1). It has been evaluated that if the realistic water-steam mixture distribution in the fuel channel [1] were used in our model, the averaged neutron energy spectrum in graphite would remain almost the same (deviations are less than 1% in thermal and epithermal region and less than 3% in fast neutron region). The share of thermal neutrons would differ about 5% in the bottom and upper part of the core from the share of the averaged neutron flux in graphite. However, such differences do not have much influence on the real axial distribution of the flux in the reactor, where the flux of thermal neutrons is regulated by means of the control rods and may be neglected in modeling. As far as the erbium poison is concerned, the fuel in model III has been replaced with the erbium fuel of 13 MWd/kgU burnup. In this case the difference in the neutron energy spectrum in graphite is less than 4%. Taking into account the fact that the erbium fuel campaign in the Ignalina NPP Unit 1 constitutes less than 1/3 of the whole operation period, this difference may not be regarded as significant in our calculations.

Activation calculations

The conservative values of measured impurities concentrations (Table 2) were used for the transmutation calculations with CINDER'90. The remaining values of impurity concentrations were taken from the scientific literature and averaged. The result of the calculations of the activity of graphite under irradiation conditions, as described above, is the list of more than 1300 radionuclides with the range of their half-lives from 10^{-7}s to 10^{20} y. The most important radionuclides taking into account their half-lives, activity and toxicity (the relation of activity to its clearance level[1]) and their corresponding reactions are:

^{6}Li(n,α) → tritium,
^{13}C(n,γ) → ^{14}C,
^{14}N(n,p) → ^{14}C,
^{35}Cl(n,γ) → ^{36}Cl,
^{40}Ca(n,γ) → ^{41}Ca,
^{54}Fe(n,γ) → ^{55}Fe,
^{59}Co(n,γ) → ^{60}Co,
^{62}Ni(n,γ) → ^{63}Ni,
^{107}Ag(n,γ) → ^{108m}Ag,
^{133}Cs(n,γ) → ^{134}Cs,
^{151}Eu(n,γ) → ^{152}Eu,
^{153}Eu(n,γ) → ^{154}Eu,
^{154}Eu(n,γ) → ^{155}Eu.

Some important radionuclides such as ^{137}Cs, ^{90}Sr are the fission products of uranium or thorium. Moreover, the presence of uranium and thorium in graphite results in a number of transuranium isotopes, the concentrations of which, in some cases, are much higher than their clearance levels.

The total activity of isotopes in all reactor graphite constructions as a function of radial distance and the time after the shutdown of the reactor is presented in Fig. 5a. The activity is constant in the central part of the reactor core and it decreases towards the periphery of the core. However, in the side reflector at the edge of the core, the total activity increases. It is due to the

[1] The clearance level refers to the activity of individual radionuclide in a material, when the free release from such material to the environment is harmless. For the purpose of this article, the 1/10th of the IAEA clearance levels [10] was used. It corresponds to the free release from material, the weight of which exceeds 3 tons.

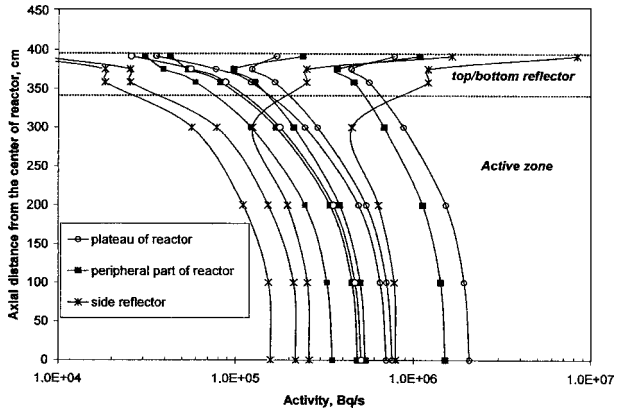

Fig. 6. Axial distribution of the activity in the RBMK-1500 reactor after 1, 30, 300 and 3000 years (from the left to the right) after shutdown of the reactor.

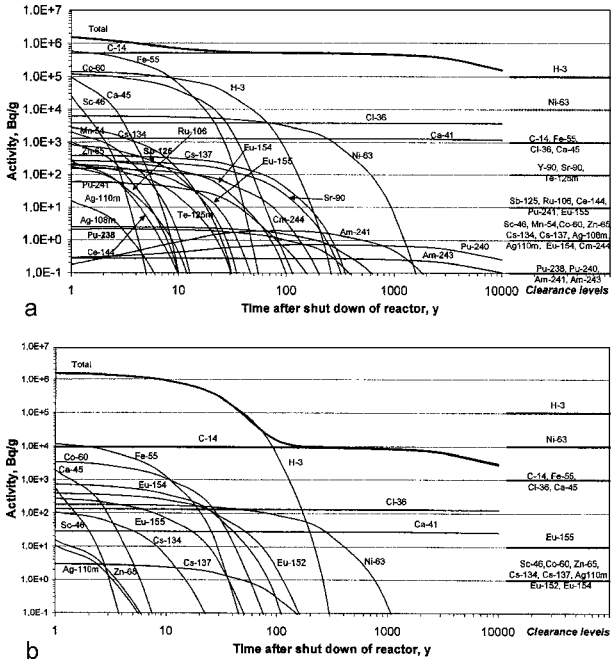

Fig. 7. Specific activities of radionuclides as a function of cooling time: a – in the moderator; b – in the corner reflector as a function of time.

higher flux of thermal neutrons in the reflector whose contribution to the total activity is essential. The total activity decreases sharply after shutdown of the reactor because of the decay of short-lived nuclides, while after 10 years of cooling it changes very slowly for a long time. Figure 5b shows the total activity in each of the graphite constructions of the reactor. At the beginning, the activity in the moderator and the corner reflector is almost equal. But the activity in the corner reflector decreases significantly in 100 years and becomes proportional to the initial neutron flux as for the rest of graphite constructions. A similar situation is observed in case of axial distribution of the total activity in the reactor (Fig. 6), but the reverse axial distribution of the total activity in the side reflector when it reaches the corner reflector area is even more representative.

The comparison of irradiated graphite radionuclide composition of moderator and corner reflectors gives the explanation for such inversion, where the radionuclide responsible for the activity in the corner reflector is tritium. Tritium is mainly produced by the reaction $^{6}Li(n,\alpha) = \alpha + {}^{3}H$. However, in moderator almost the whole ^{6}Li is burnt out within two years of the reactor operation due to its big neutron capture cross-section. It corresponds to the fluence level of approximately 10^{22} n cm^{-2}. Afterwards, only the decay of tritium defines its balance in the graphite structure. But such fluence is not reached in the corner reflector and tritium is being continuously produced during all the time of operation. The same effect is observed for other mother-nuclei with big neutron capture cross-sections when the half-lives of resulting radionuclides are comparable with the operation time of the reactor, e.g. for ^{151}Eu, ^{153}Eu. The activity of ^{152}Eu, ^{154}Eu and ^{155}Eu, which is produced by double neutron capture reaction $^{153}Eu(n,\gamma) \rightarrow {}^{154}Eu(n,\gamma) \rightarrow {}^{155}Eu$, is higher in the corner reflector than in the moderator.

The specific activities of radionuclides, which exceed their clearance levels in the given graphite structure, as well as other radionuclides which may be important from radiological point of view, in moderator and in the corner reflector, are shown in Fig. 7. In the right part of the figure the clearance levels for individual radionuclides are presented. Although ^{14}C and tritium constitute major activity in all the irradiated graphite constructions, respectively 7.03×10^{14} Bq and 3.83×10^{14} Bq, and are crucial to the long-term waste management, the role of other radionuclides such as ^{55}Fe and ^{60}Co, etc. has to be duly evaluated, especially for dismantling and on-site waste management purposes. The presence of transuranium elements in the quantities shown in Fig. 7 may not cause particular risk to waste handling. However, their average activity in the moderator is about 230 Bq/g, which is rather close to the 370 Bq/g limit recommended by the IAEA for acceptance of waste packages [15] and is likely to be exceeded because of eventual uncertainties due to the initial impurity of U and irradiation parameters.

Estimation of uncertainties

The accuracy of such activation calculations is determined, mainly, by the uncertainties in the effective neutron cross-section data libraries used and by the precision of data on impurity concentrations. The uncertainty of impurity concentrations, as reported in [8], prevails. The presence of impurities is due to the raw materials involved in fabrication of nuclear graphite. Therefore, their concentrations are sensitive to technological process and sometimes vary even for the same grade of graphite. The influence of such variations on the results of calculations of the total activity is shown in Fig. 8. The use of maximal key-element concentrations gives the total activity which is by an order of magnitude higher than the total activity in the case of their minimal concentrations. Only one key-isotope, namely natural ^{13}C, the concentration of which in carbon is about 1.1%, is practically invariant. Estimations show that for the neutron flux of the RBMK-1500 reactor

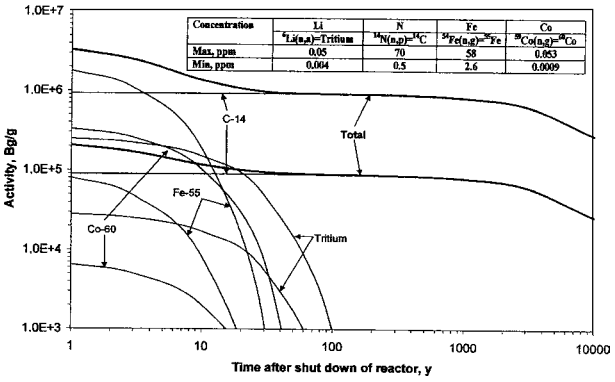

Fig. 8. Calculated total activity in the moderator in the case of maximal and minimal concentration of impurities of key-elements as a function of cooling time. The impurity data are taken from [6, 7, 16].

the production rate of ^{14}C from ^{13}C is equal to the production rate from ^{14}N if the concentration of ^{14}N is about 7 ppm. In fact, the production rate of a radionuclide and the neutron reaction rate per unit of flux may differ from one graphite construction to another due to the difference in spectrum of the neutron flux in these constructions (Fig. 3). Table 3 shows the neutron reaction rate in thermal, epithermal and fast neutron energy regions. Although the major activity is a result of reactions on thermal neutrons, some individual radionuclides as ^{60}Co, ^{108m}Ag, ^{134}Cs or ^{154}Eu are produced, in comparable quantities, by reactions with epithermal neutrons. Therefore, activity calculations of these radionuclides based only on thermal neutrons may lead to a considerable underestimation of their total activity.

The uncertainties, which are the result of the use of different effective neutron cross-section data libraries, are less important than those caused by uncertainties of impurity concentrations. Nevertheless, for some individual radionuclides they are not negligible. The best known nuclear data libraries today, ENDF/B-VI, JEF, and JENDL, were compared with the original CINDER'90 library. Table 4 gives differences of these libraries calculated for the neutron flux in the RBMK-1500 moderator for thermal, epithermal and fast neutrons.

Combining the data from Tables 3 and 4 one may conclude that significant variation of results for some individual radionuclides may take place because of the use of a certain cross-section data library and a negligible influence of epithermal neutrons. For the above estimations the following neutron energy intervals were taken: thermal (10^{-3} eV – 0.7 eV), epithermal (0.7 eV – 0.1 MeV), fast (0.1 MeV – 25 MeV).

Table 3. Relative production rates of some important radionuclides in different neutron energy regions within the moderator

Reaction	Neutron energy region		
	Thermal (55.0%)	Epithermal (32.68%)	Fast (11.82%)
$^{6}Li(n,\alpha) \rightarrow$ tritium	96.47%	3.50%	0.03%
$^{13}C(n,\gamma) \rightarrow {}^{14}C$	87.49%	3.39%	9.12%
$^{14}N(n,p) \rightarrow {}^{14}C$	95.92%	3.48%	0.60%
$^{35}Cl(n,\gamma) \rightarrow {}^{36}Cl$	96.84%	3.16%	0.00%
$^{40}Ca(n,\gamma) \rightarrow {}^{41}Ca$	95.82%	4.01%	0.17%
$^{54}Fe(n,\gamma) \rightarrow {}^{55}Fe$	95.68%	4.19%	0.13%
$^{59}Co(n,\gamma) \rightarrow {}^{60}Co$	84.63%	15.36%	0.01%
$^{62}Ni(n,\gamma) \rightarrow {}^{63}Ni$	96.74%	3.23%	0.03%
$^{107}Ag(n,\gamma) \rightarrow {}^{108m}Ag$	77.53%	22.16%	0.31%
$^{133}Cs(n,\gamma) \rightarrow {}^{134}Cs$	47.50%	52.43%	0.07%
$^{151}Eu(n,\gamma) \rightarrow {}^{152}Eu$	97.62%	2.38%	0.00%
$^{153}Eu(n,\gamma) \rightarrow {}^{154}Eu$	65.19%	34.75%	0.06%
$^{154}Eu(n,\gamma) \rightarrow {}^{155}Eu$	97.30%	2.69%	0.01%

Table 4. Differences of neutron effective cross-sections using different libraries (related to ENDF/B-VI or to the library with the note "Ref.")

Reaction	Thermal region			Epithermal and fast region		
	JEF	JENDL	CINDER'90	JEF	JENDL	CINDER'90
$^{6}Li(n,\alpha) \rightarrow$ tritium	–0.164%	0.106%	–0.580%	–0.064%	0.119%	–0.757%
$^{13}C(n,\gamma) \rightarrow {}^{14}C$	–	–	–32.00%	–	–	–74.99%
$^{14}N(n,p) \rightarrow {}^{14}C$	–0.558%	–2.995%	95.58%	–1.689%	–4.822%	92.78%
$^{35}Cl(n,\gamma) \rightarrow {}^{36}Cl$	–	Ref.	–0.170%	–	Ref.	2.674%
$^{40}Ca(n,\gamma) \rightarrow {}^{41}Ca$	–	Ref.	0.057%	–	Ref.	0.000%
$^{54}Fe(n,\gamma) \rightarrow {}^{55}Fe$	14.80%	–4.17%	0.126%	15.81%	12.54%	0.117%
$^{59}Co(n,\gamma) \rightarrow {}^{60}Co$	–0.154%	0.111%	0.430%	–0.475%	–0.335%	–1.868%
$^{62}Ni(n,\gamma) \rightarrow {}^{63}Ni$	–0.178%	–1.218%	0.583%	–0.090%	16.89%	0.325%
$^{133}Cs(n,\gamma) \rightarrow {}^{134}Cs$	–1.726%	–1.822%	2.323%	14.41%	3.347%	2.155%
$^{151}Eu(n,\gamma) \rightarrow {}^{152}Eu$	–0.186%	0.319%	–31.21%	–0.313%	–10.56%	–34.53%
$^{153}Eu(n,\gamma) \rightarrow {}^{154}Eu$	–15.85%	Ref.	4.659%	3.036%	Ref.	20.631%
$^{154}Eu(n,\gamma) \rightarrow {}^{155}Eu$	–79.13%	Ref.	–16.99%	119.62%	Ref.	11.05%

Conclusions

The radioactivity of the Ignalina NPP RBMK-1500 reactor graphite constructions was calculated for the first time taking into account the power load history of Unit 1, measured impurities of graphite used in the Ignalina NPP. The calculations were based on computer codes MCNPX and CINDER'90, which were validated for graphite reactors in earlier studies. The modeling procedure involved both neutron fluxes and their total energy spectrum in graphite constructions. It has been found that the spectrum of neutrons in graphite is not influenced significantly by burnup of the fuel or by the position of control rods. Main radionuclides, which contribute to the total radioactivity, are ^{14}C, tritium, ^{60}Co and ^{55}Fe. However, their share differs from one graphite construction to another, especially for tritium, as an effect of the burnup of its mother nucleus – Li. The principal uncertainty of radioactivity calculations is due to a high variation of impurity concentrations in nuclear graphite. Although the major activity is created by nuclear reactions with thermal neutrons, some radionuclides relevant to radiation protection are produced by epithermal neutrons as well. To avoid underestimation of these radionuclides, the whole neutron energy spectrum and its variation in different graphite constructions must be taken into account. Comparison of main neutron effective cross-section libraries for the typical neutron flux in the RBMK-1500 graphite shows significant uncertainties for some radionuclides, which implies the need of their update.

Acknowledgment We gratefully acknowledge the French Commission of Atomic Energy (CEA) for the fellowship grant and the partial funding of this work. One of the authors (D.A.) would like to thank the colleagues of CEA Saclay (DSM/DAPNIA/SPhN) for their support and pleasant environment during his scientific visit.
This research was also supported by the Lithuanian State Science and Studies Foundation under the project No. C-19/2004 and Franco-Lithuanian project "Gilbert".

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