BULLETIN OF THE POLISH ACADEMY OF SCIENCES
TECHNICAL SCIENCES, Vol. 61, No. 1, 2013
DOI: 10.2478/bpasts-2013-0015

CIVIL ENGINEERING

Monte-Carlo aided design of neutron shielding concretes

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Abstract. The process of design of building composites, like concrete is a complex one and involves many aspects like physical and mechanical properties, durability, shielding efficiency, costs of production and dismantlement etc. There are plenty of parameters to optimize and computer tools can help to choose the best solution. A computer aided design plays an important role nowadays. It becomes more accurate, faster and cheaper, so laboratories often apply computer simulation methods prior to field testing. In case of nuclear engineering, the radiation shielding problems are of much importance, because safety of such facilities is a key point. In this article the most effective methods for neutron shielding studies based on Monte-Carlo simulations of neutron transport and nuclide activation studies in concrete are presented. Two codes: MCNPX and CINDER'90 are extensively used to compare the shielding efficiency of commonly used concretes and to study the influence of concentration of B, Ba and Fe elements on shielding efficiency.

Key words: nuclear, neutron, shielding, concrete, Monte Carlo Simulations, MCNPX, CINDER'90.

1. Introduction

Neutron radiation, which is present in nuclear reactor facilities is hard to attenuate. Neutrons have no electrical charge and can be imagined as highly penetrating balls bouncing on atoms of a shielding material. Probabilities of nuclear reactions strongly depend on neutron energy but the most probable interaction of such a ball with a matter is an elastic scattering. This and other behaviours are evaluated as nuclear data and presented as cross-section tables used in simulation codes.

For neutron shielding problems, the Boltzmann transport equation for neutrons should be resolved. Each term represents a gain or a loss of a neutron, and the balance, in essence, claims that neutrons gained equals neutrons lost. The particular symbols can be marked differently depending on the author.

$$\left[\frac{1}{\nu(E)}\frac{\partial}{\partial t}+\widehat{\Omega}\cdot\nabla+\Sigma_t(r,E,t)\right]\psi(r,E,\widehat{\Omega},t)$$
$$=\frac{\chi_p(E)}{4\pi}\int_0^\infty dE'^{\nu_p}(E')\Sigma_f(r,E',t)\Phi(r,E',t)$$
$$+\sum_{i=1}^{N}\frac{\chi_{di}(E)}{4\pi}\lambda_i C_i(r,t)$$
$$+\int_{4\pi}d\Omega'\int_0^\infty dE'^{\Sigma_S}(r,E'\to E,\widehat{\Omega}'\to\widehat{\Omega}',t)$$
$$\cdot\psi(r,E',\widehat{\Omega}',t)+s(r,E,\widehat{\Omega},t).$$

The Boltzmann transport equation cannot be solved analytically unless lots of simplifying assumptions are made. To obtain physically realistic solutions to the transport equation there is a need to use numerical techniques.

One of the simplest method for that is a use of the computer Monte-Carlo (MC) methods developed by Stanislaw Ulam, John von Neumann and Nicholas Metropolis [1] during their work in Los Alamos laboratory. The MC method is a method of multiple sampling of a neutron history. Every simulation consists of a big number of repeated single neutron histories and the final result is obtained statistically. Besides the MC method, there are other possibilities for a neutron transport calculation like the discrete ordinates method (codes ANISN [2] and DOT [3]), the remove-diffusion method (code SAMSY [4]). For simple shielding calculations linear attenuation coefficients can also be used [5], but the method giving the most reliable results is the MC one (codes MORSE [6] and MCNPX [7]). The basic difference is that a discrete ordinate method simulates the mean neutron flux, going through some elemental geometry cells, whether the Monte Carlo method simulates multiple times the history of single neutron jumping straight line between events (collisions) in the shielding material and from these multiple histories desired statistical quantities are obtained.

The MC method has one important drawback. As it is a statistical method it strongly depends on amount of sampling. The uncertainty of MC calculation is proportional to $1/\sqrt{N}$ and depending on problem, fair results need a lot of real time simulation. It is the main reason that other methods are also used mainly for time critical operations.

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The basis of the MC method is generally a Random Number Generator (RNG) that provides sampling of neutron initial energy, direction of motion, step length, interacting nucleus, type of interaction, new direction etc. Essential information which makes the simulation close to the reality is the library of cross-section nuclear data. In case of MCNPX, the standard library, also used in this work is ENDF/B VI. RNG sample the provided experimentally evaluated cross-sections for different reactions and finally the statistical result is available.

A simplified algorithm of the MC neutron transport simulation is presented in Fig. 1. First (1) neutron is generated. It depends on source configuration:

- if it is monoenergetic neutrons source, the energy is constant otherwise it is sampled from the provided distribution by RNG (ex. in a histogram manner);
- if it is the point source, surface or volume source etc, so an initial position may be tossed from some provided distributions by RNG;
- if it is monodirectional or a direction is sampled from some provided distribution by RNG.

After this initial work, a neutron (if it is present in a defined material) moves in a straight line for the sampled length (2) obtained from:

$$l = -\frac{1}{\Sigma_t} ln(\xi),$$

where ξ is sampled by RNG from an uniform distribution [0,1). Σ_t – is a total macroscopic cross section of a material, which is a sum of macroscopic cross-sections for each element

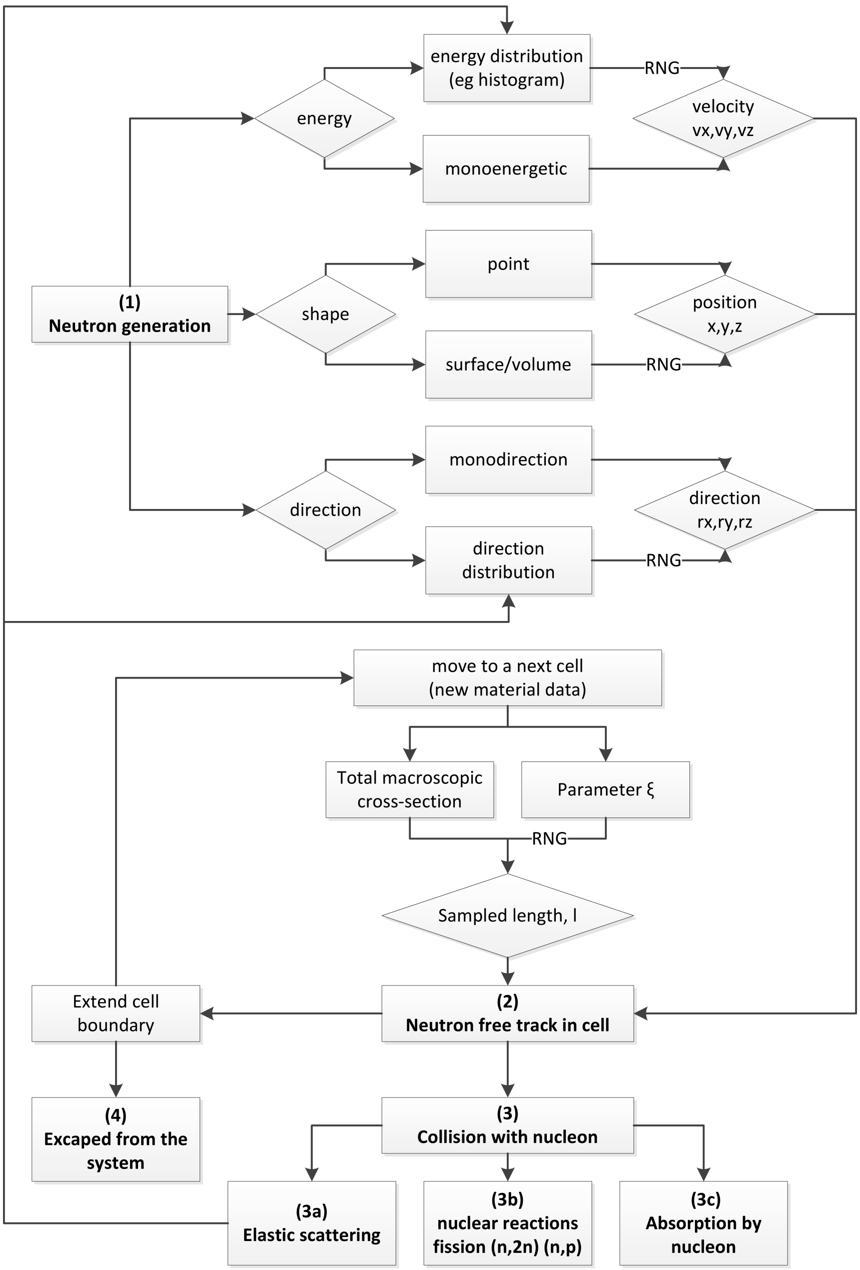

Fig. 1. Simplified algorithm of MC neutron transport simulation

in material. The macroscopic cross-section is a multiplication of the microscopic cross-section by a concentration of an element in the material.

When the new neutron position is calculated and it is within the call boundary range, it becomes a collision point (3), where a collision element is selected by tossing from an uniform distribution by RNG which probability is given from the macroscopic cross-section of an element divided by a total macroscopic cross-section of a cell. Next by the same manner a type of collision is selected. If an elastic scattering is chosen (3a) then a new direction is sampled by RNG using the differential angular cross-sections and on behalf of energy and momentum conservation laws a new velocity is calculated. Then a new neutron history begins – back to (2). If neutron crosses the boundary of the cell, then its contribution can be included in a surface detector. If neutron escapes the whole simulation system – neutron leakage (4) or an absorption (3c) as an interaction is selected in a collision point, a new history should begin. There could be also a plenty of other interactions from inelastic scattering which introduce new particle into simulation like neutrons, protons, deuterium, tritium, helium and more heavy ions. Fissile nuclides can also undergo the fission process, which also produce prompt and delayed neutrons. During the collision a photon emission is also simulated. Because lot of a photon production and their negligible influence, they are discarded by the Russian roulette method when appropriate.

The neutron shielding design is governed by the term: ALARA (As Low As Reasonably Achievable) which is commonly used in the nuclear industry. Shielding should be optimized for size and weight, because in case of heavy weight shielding, a proper construction support must be prepared. Generally, neutron attenuation efficiency should be high and activation should be low. Production costs also should be low, because shields usually consume lots of material. Different materials are considered for neutrons shields like parafine, boron carbides [8], zirconium borohydrides or zirconium hydrides [9], resin based materials [10–12]. They differ in many physical properties, for example some of them are heat resistant. Concretes which have excellent cost ratio are main shielding material, but their shielding efficiency vary on material composition and locally produced concretes often differ, so it is very important to carry out research on them. Such work is presented in [13, 14]. Lately an intensive development of Polymer-Cement Concrete (PCC) in the last decade has been observed [15]. It opens a new research are on influence of polymer additions to concrete on shielding properties by an increase of desirable hydrogen content in composite mass [16].

2. Simulation set up for LWR reactor neutron flux

The Monte Carlo neutron transport simulation was performed with MCNPX 2.5.0 [7]. The geometry of sample was a simple slab with dimensions 1 m × 1 m × 2 m. It was divided into 8 cells of 25 cm each. A neutron source was placed in the middle of one of the smaller surfaces in the distance of 1 cm from surface. The neutron source was monodirectional (neutrons direction transversal to near surface) and had a diameter of 1 cm (neutron initial position sampled from flat distribution). The neutron current and flux needed for dose calculations was sampled on every surface which was transversal to neutron initial direction. In the sample slab 5 point detectors were also placed to calculate the flux needed in CINDER’90 [17] simulation. For flux calculations, proper energy groups were included to satisfy further calculation needs. Every simulation was performed for $1{\cdot}10^7$ neutron histories to get fair statistical properties. For monoenergetic (thermal, epithermal, fast) neutrons attenuation study $1{\cdot}10^8$ neutron histories were used. The chosen number of histories guarantee the repeatability of studies in given conditions (intro data) and it makes simulation finite in reasonable time. No variance reduction techniques were applied. Material isotopic composition and artificial additions were carefully calculated using Excel. Analysis of simulation output files was performed automatically by using Python own written scripts. Neutron source spectrum was typical LWR spectrum with total flux of $4.64{\cdot}10^9$ n/cm^2/s and was calculated with APOLLO2 code [18]. The shape of neutron spectrum is presented in Fig. 2.

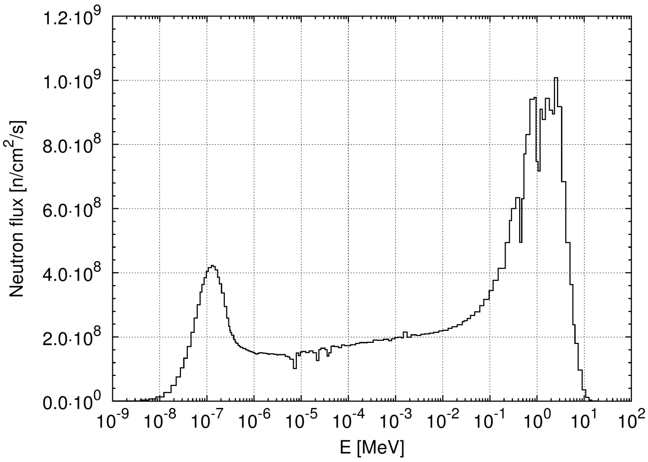

Fig. 2. Neutron source spectrum from typical LWR reactor

CINDER’90 [17] transmutation code allows for a simulation which involves decay chain calculations – production and destruction of radionuclides. For an input file, flux with fixed energy groups calculated in MCNPX – approach similar to presented in [19, 20], material initial isotopic composition and irradiation time profile was applied. As a result isotopic composition and activity of material at specified time points was obtained. For these simulations a scenario of 20 years neutron irradiation with constant flux was used. After this, during the cooling period activation data was accumulated after 1, 2, 5, 10, 20, 50 and 100 years. The neutron fluxes for these calculations were obtained from MCNPX simulation.

Table 1 and 2 shows the material composition for shield efficiency study in this article. Only the most important elements (> 1000 ppm) were chosen for simulation purposes. The same elements were used for all studied concretes.

Amount of each element was scaled accordingly to maintain the full composition. These data was provided by DSM/IRFU LENAC laboratory from real concrete chemical composition study. The mass density was not exactly known for every type of these concretes, but for comparison purposes the similar concretes have the same density.

The abbreviation used are: OC – ordinary concrete, BC – borated concrete, HC – heavy concrete, BHC – borated heavy concrete, RC – reinforced concrete. “U” and “R” designators mean Ulysse or RUS (Réacteur Universitaire de Strasburg).

Table 1
Concrete mass composition of Ulysse de Saclay decommissioned reactor (Ulysse). Units in ppm

Concrete type:	OC-U	BC-U	HC-U	BHC-U
Density g/cm^3:	2.63	2.58	3.20	3.20
H	3502	3505	1114	1119
B	16	20042	6	21983
C	37190	37218	18553	18623
O	529086	514765	306846	285852
F	0	0	51559	51756
Na	920	921	487	489
Mg	2776	2779	793	796
Al	14563	14574	4751	4769
Si	217718	211824	71586	68227
P	262	262	95	96
S	3402	3404	92899	93253
K	5153	5157	3459	3472
Ca	174094	174224	109248	109665
Ti	961	961	197	198
Mn	252	252	1102	1107
Fe	9645	9652	2025	2033
Sr	328	328	9557	9593
Ba	132	132	325722	326972

Table 2
Concrete mass composition of Strasburg University decommissioned reactor (RUS). Units in ppm

Concrete type:	OC-R	BC-R	HC-R	RC-R
Density g/cm^3:	2.63	2.58	3.20	4.42
H	1648	8851	1180	2498
B	22	18314	5	5167
C	25168	27411	2499	7734
O	512453	520201	282393	146713
F	0	0	58286	0
Na	6406	5631	574	1589
Mg	2083	7652	1216	2159
Al	24083	22180	2589	6258
Si	278133	210148	40100	59296
P	385	348	44	99
S	4441	2513	105975	709
K	4096	9103	1050	2570
Ca	114096	149616	73982	43065
Ti	948	821	120	232
Mn	495	1599	49	451
Fe	11350	13134	1440	720663
Sr	671	1645	10597	553
Ba	13522	834	417901	244

3. Concrete shielding and activation from LWR reactor neutron flux

Neutron attenuation in different concrete types of both places (Ulysse and RUS) are presented in Figs. 3 and 4. These data were calculated on basis of neutron current and normalized to 1 at the entry surface of concrete slab. The best neutron attenuation properties are present in borated concretes. Borated and heavy concrete differ by more than an order of magnitude in favour of borated concrete in case of Ulysse concretes and almost two orders of magnitude for RUS concretes. For neutron attenuation the most important nuclides in shields are the smallest ones like hydrogen, as the neutron elastic scattering is the most effective with them (the most amount of energy is transferred per collision). These simulations confirm that the best neutron shields are made from borated concretes, because ^{10}B is a very good neutron absorber (its natural abundance is 19.9%) and it is quite light element. The heavy concretes are prepared to act as a shield for gamma radiation because of increased concentration of heavy elements which have rich electron shells.

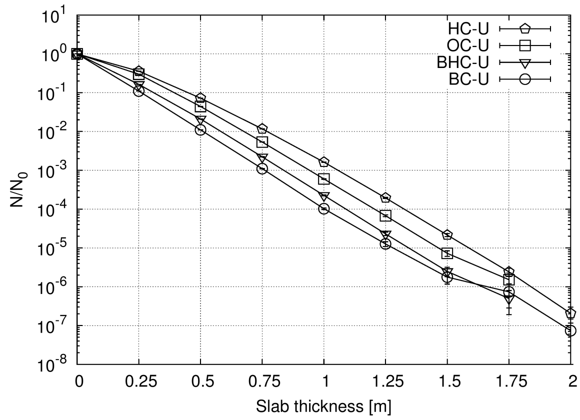

Fig. 3. Neutron attenuation in Ulysse concretes

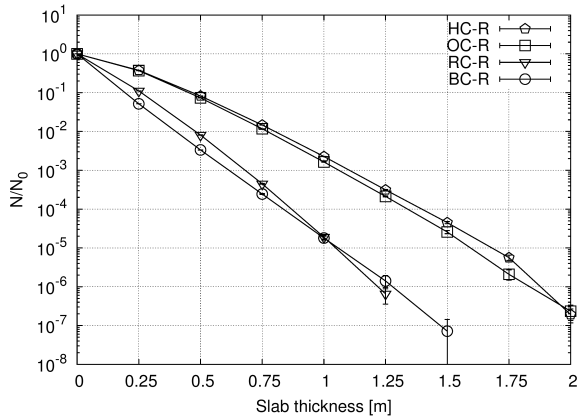

Fig. 4. Neutron attenuation in RUS concretes

In Figs. 5 and 6 there are results of the simulation of 20 years activation of materials in depth of 1 cm and after 1 and more years of cooling. The least activated materials were heavy concretes. Borated concretes remain activated for a long time. In the case of reinforced concretes, the activation for the first 10 years was very high but after that period of time went down very fast to the level of the best ones.

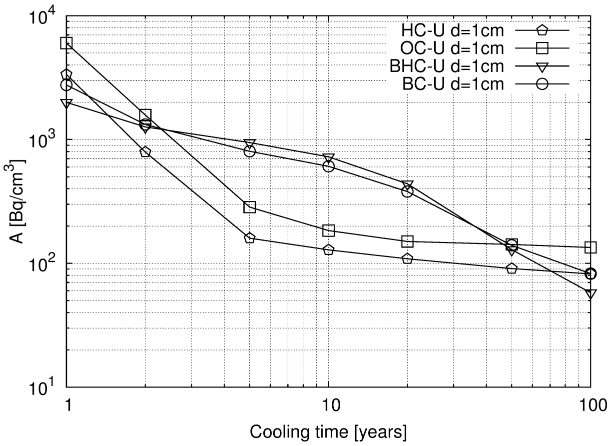

Fig. 5. Total activation of Ulysse concretes after 20 years of neutron irradiation

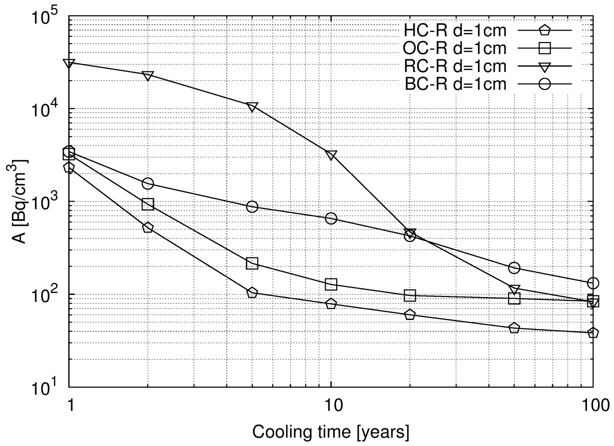

Fig. 6. Total activation of RUS concretes after 20 years of neutron irradiation

Table 3
The most active nuclides in Ulysse concretes. After 20 years of irradiation and 1 year of cooling

OC-U	A [Bq/cm^3]	BC-U	A [Bq/cm^3]	HC-U	A [Bq/cm^3]	BHC-U	A [Bq/cm^3]
Ca 45	5442.67	Ca 45	1660.76	Ca 45	2735.51	H 3	1090.37
Fe 55	338.34	H 3	835.55	S 35	377.93	Ca 45	689.76
Ca 41	74.57	Fe 55	104.62	Fe 55	57.14	S 35	91.60
Ar 39	71.50	Ar 39	69.31	Ar 39	54.14	Ar 39	54.85
Ar 37	59.60	Ar 37	53.02	H 3	42.72	Ar 37	37.44
S 35	17.23	Ca 41	23.01	Ar 37	40.14	Fe 55	14.73
Mn 54	6.25	Mn 54	6.07	Ca 41	37.65	Ca 41	9.70
C 14	3.95	S 35	5.15	Ba133	6.84	Mn 54	3.49
H 3	1.04	C 14	2.19	Mn 54	3.51	Ba133	2.56
K 40	0.39	K 40	0.39	C 14	2.11	C 14	1.19
Na 22	0.19	Na 22	0.17	K 40	0.32	K 40	0.32
Cl 36	0.09	Cl 36	0.06	Na 22	0.13	Na 22	0.13
Fe 59	0.04	Fe 59	0.01	Cl 36	0.05	Cl 36	0.04
Sc 46	0.01	Sc 46	0.01	Sr 85	0.05	Sr 85	0.03

Table 4
The most active nuclides in RUS concretes. After 20 years of irradiation and 1 year of cooling

OC-R	A [Bq/cm^3]	BC-R	A [Bq/cm^3]	HC-R	A [Bq/cm^3]	RC-R	A [Bq/cm^3]
Ca 45	2769.84	Ca 45	2206.73	Ca 45	1753.16	Fe 55	28444.36
Fe 55	308.85	H 3	796.77	S 35	406.26	Ca 45	1506.66
Ar 39	54.98	Fe 55	221.09	H 3	48.00	Mn 54	955.67
Ca 41	37.95	Ar 39	121.98	Fe 55	38.23	H 3	425.85
Ar 37	36.74	Ar 37	49.00	Ar 37	25.47	Ar 39	75.68
S 35	17.42	Ca 41	30.66	Ca 41	24.02	Ar 37	28.86
Mn 54	7.40	Mn 54	9.85	Ar 39	15.75	Ca 41	20.72
C 14	3.26	S 35	5.95	Ba133	8.33	S 35	3.99
H 3	1.28	C 14	2.63	C 14	1.82	Fe 59	3.72
Na 22	1.20	Na 22	1.03	Mn 54	1.03	C 14	1.48
K 40	0.31	K 40	0.67	Na 22	0.16	Na 22	0.72
Ba133	0.27	Cl 36	0.12	K 40	0.10	K 40	0.33
Cl 36	0.06	Fe 59	0.03	Sr 85	0.05	Cl 36	0.07
Fe 59	0.04	Ba133	0.01	Cl 36	0.02	Sc 46	0.01

Most active isotopes produced in studied concretes during 20 years of irradiation and 1 year of cooling are presented in Tables 3 and 4. In most of concretes (excluding reinforced concrete), the most active element is ^{45}Ca. In case of reinforced concrete (RC-R) the most active element ^{55}Fe takes main part of total activation with over then 28 kBq/cm^3 after 1 year of cooling. Many of these products like H, He, Ar take gaseous form and normally can leave the concrete slab during the irradiation process. Material which is good neutron absorber usually will get higher activated, so it is a compromise between good neutron attenuation efficiency and material activation level and of course how long material remains active. It is also very important what type of active products are present in material. Some of them for example may decay with high energy gamma emission, which has appropriately higher impact on effective dose. In paper [21] it is presented that most important radionuclides, which accumulates in concretes are ^{60}Co, ^{152}Eu and ^{154}Eu. In our simulation for ordinary concrete only ^{60}Co exists immediately after irradiation period, but in very small amount with activity of about $1.78 \cdot 10^{-5}$ Bq/cm^3, which not exceed the clearance levels established by IAEA [22].

In Fig. 7, there is an isotopic concentration (number of elements per cubic meter) before irradiation (normal line) and after 20 years of irradiation and 1 year of cooling (dashed line). Broad number of nuclides production was observed.

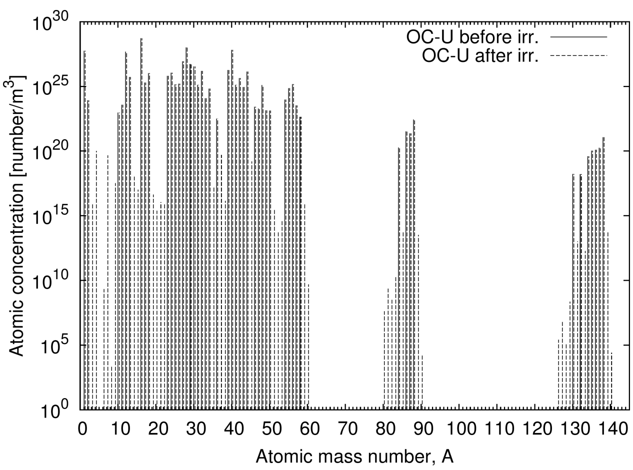

Fig. 7. Composition of material before and after 20 years of neutron irradiation and 1 year of cooling

In Figs. 8 and 9 there are effective dose rates calculated for hypothetical human body behind the shield wall. Doses were calculated with factors obtained from [23] with the use of neutron and secondary gamma production in material due to nuclear reactions. Generally, it is clear that the borated concretes are better shielding than others. An interesting fact is that heavy-weight concretes are not much better than ordinary ones – the cost of producing them is not very effective. The best solution seems to be reinforced concrete – 1.25 m thickness provides almost 100% shielding.

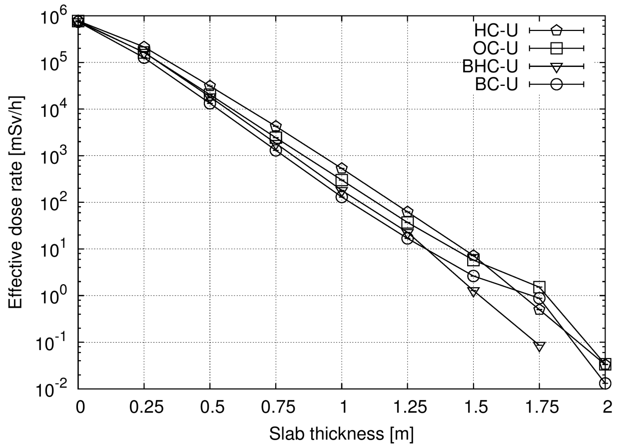

Fig. 8. Effective dose rate, calculated for human standing behind the slab. Ulysse concretes

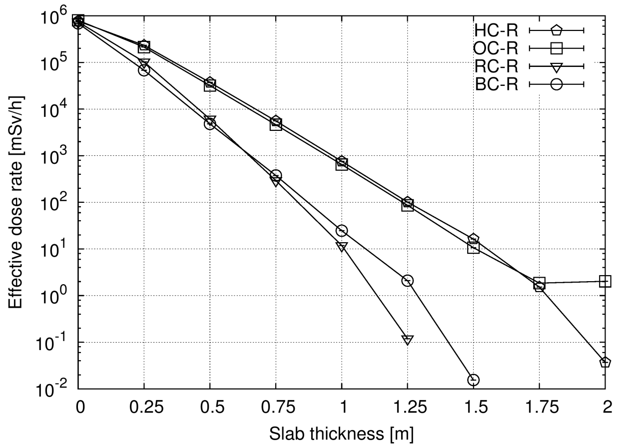

Fig. 9. Effective dose rate, calculated for human standing behind the slab. RUS concretes

4. Modified concrete shielding and activation from LWR reactor neutron flux

For a neutron shielding material many parameters should be known. They are for example how material composition will affect on attenuation, activation level and how long high level of activation will remain. These information become more important when nuclear facility dismantlement is planned.

Influence of B, Ba and Fe elements were studied in artificially modified material. Initial composition was OC-U (Ordinary Concrete from the Ulysse). When element concentration was changed, all others were scaled appropriate to maintain the full composition. Mass density of material has much influence on shielding efficiency, so mass density was also changed by linear interpolation method between known densities of concretes with known amount of investigated additive. For example in OC-U there is 9645 ppm of Fe and mass density of this concrete material is 2.63 g/cm^3, and in RC-R there is 720663 ppm of Fe with mass density of 4.42 g/cm^3. A line between these 2 points is governed by:

$$d(c_{Fe}) = 2.52 \cdot 10^{-6} * c_{Fe} + 2.61,$$

where c_{Fe} is Fe concentration in ppm and d is the mass density in g/cm^3.

Influence of B, Fe and Ba concentration on neutron attenuation efficiency is presented in Fig. 10. Increased B addition makes the shield more effective. The same behaviour is observed in case of Fe up to $5 \cdot 10^5$ ppm. Increasing the Fe amount more make the shield worse probably because of gradually decreased amount of light elements like H. The same behaviour is observed in case of high amount of Ba element. One more conclusion is that changing amount of Ba in wide area has no influence on shielding efficiency. It is in agreement with preliminary studies made by Piotrowski et al. [24].

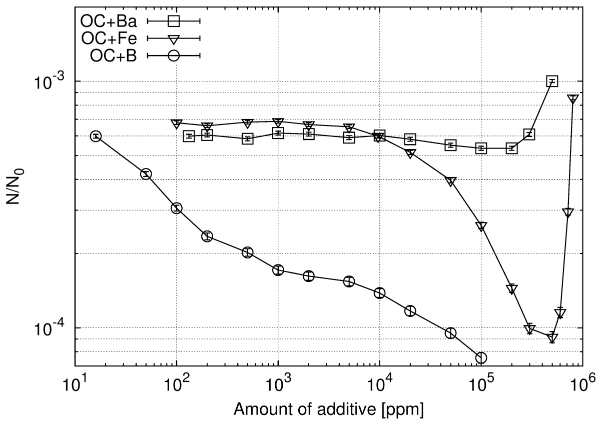

Fig. 10. Influence of B, Ba and Fe additives on neutron attenuation in material, $d = 1.0$ m

Activation levels of materials are presented in Figs. 11–17 consequently for cooling times of 1, 2, 5, 10, 20, 50, 100 years. It is clear that the most activated concretes had a large amount of iron. The least active concretes on the cooling begin were these with boron addition. After 10 years of cooling the situation changes, concretes with high concentration of B is more active than these with large amount of Ba and Fe. Later up to 50 years concrete with Fe is the least active. After 100 years of cooling an activation level for all concretes has been almost the same at values of about 102 Bq/cm^3. It means that this period is sufficient for cooling all types of concrete elements before decommissioning and recycling. If the time needs to be shorter the specific analyse should be done.

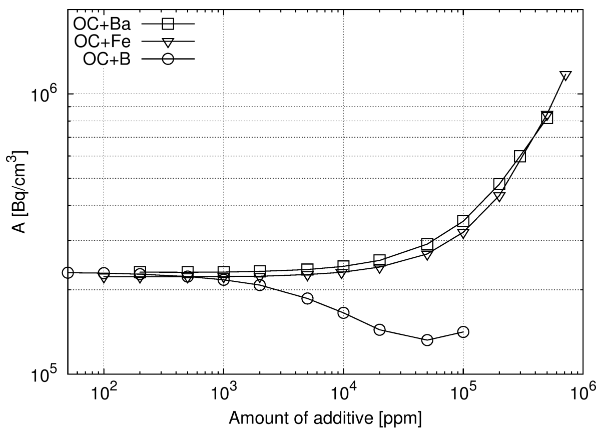

Fig. 11. Activation of material after 20 years of neutron irradiation at constant flux. No cooling

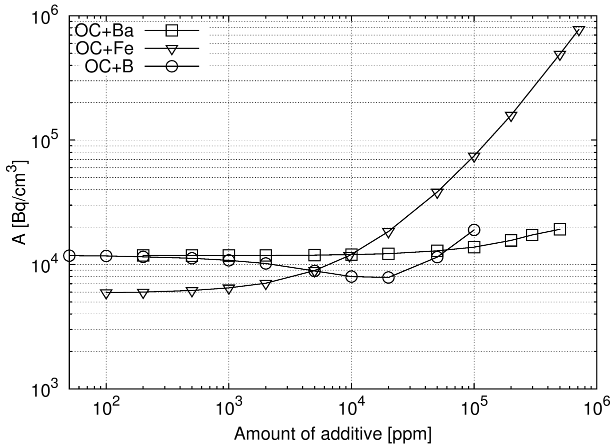

Fig. 12. Activation of material after 20 years of neutron irradiation at constant flux. 1 year of cooling

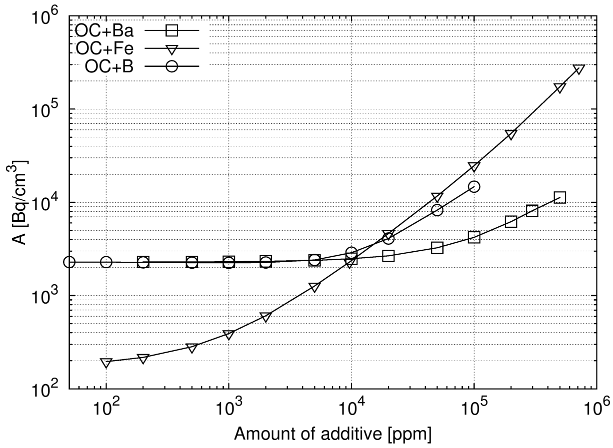

Fig. 13. Activation of material after 20 years of neutron irradiation at constant flux. 5 years of cooling

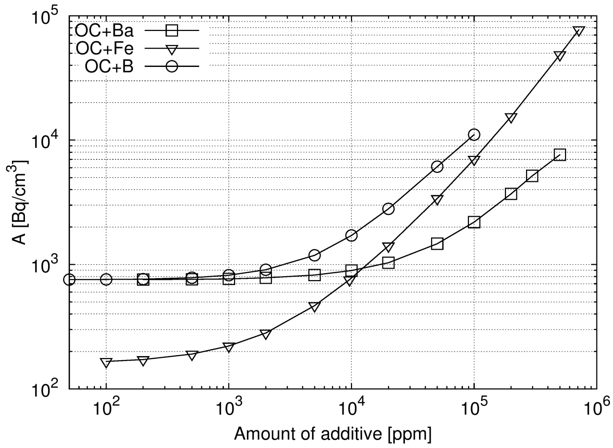

Fig. 14. Activation of material after 20 years of neutron irradiation at constant flux. 10 years of cooling

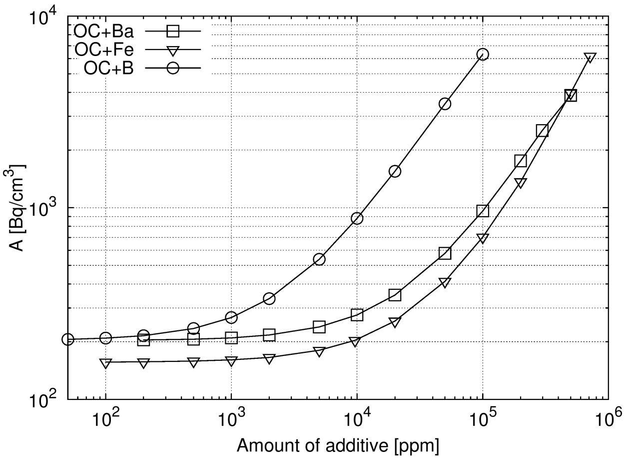

Fig. 15. Activation of material after 20 years of neutron irradiation at constant flux. 20 years of cooling

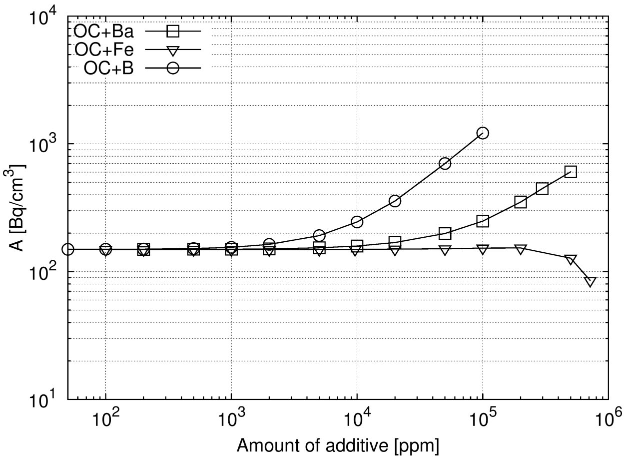

Fig. 16. Activation of material after 20 years of neutron irradiation at constant flux. 50 years of cooling

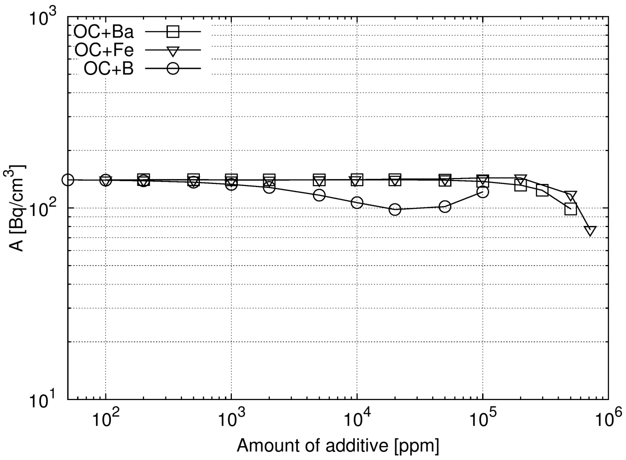

Fig. 17. Activation of material after 20 years of neutron irradiation at constant flux. 100 years of cooling

5. Mono energetic neutron radiation study of concrete shielding efficiency

In accelerator facilities (suited for medical purposes or scientific research) a much more energetic neutrons may be present. Also future fusion reactors will need neutron shields (typical 2.5 MeV energy for D+D reaction or even 14 MeV for D+T reaction). High energy neutrons are even harder to attenuate [25, 26] and nuclides cross-sections for inelastic reactions like (n,2n) are higher, so reactions with neutron multiplication occurs. There are also more other particle generated like p, d, t, ^{3}He, α, and more heavy ions. For study of monoenergetic neutrons attenuation in materials, $1\cdot10^8$ neutron histories were used. The results as a attenuation depending on the slab thickness are presented in Figs. 18–25. According to simulation, thermal neutrons (E = 0.025 eV) didn't pass through 25 cm of Ulysse borated concrete (BC-U), Ulysse borated heavy concrete (BHC-U), RUS borated concrete (BC-R) and RUS reinforced concrete (RC-R). They were also well attenuated in Ulysse ordinary concrete (OC-U). It confirms that in thermal neutron facilities the neutron shielding is not the main problem for safety assurance. Increasing energy to 2.5 MeV and 10 MeV allows for drawing some conclusions about the shielding efficiency. Generally, there is no difference between ordinary and heavy-weight concrete. The change could be made by introducing into a mix the boron addition or reinforcing. For 100 MeV neutrons a strong neutron multiplication occurred in all types of concrete without addition of boron in first 75 cm region. From this comparison an Ulysse borated heavy concrete and RUS reinforced concrete seems the best neutron shielding material. Generally concrete materials with addition of neutron absorbing elements like boron or iron (in reinforced concrete) perform better in all neutron spectrum. A high concentration of hydrogen is also important because it is the most efficient neutron moderator. The worst materials were full of high mass elements of very small cross sections for neutron capture like Ba – it confirms that heavy-weight concrete is no necessary material for shielding for neutron emitting facilities.

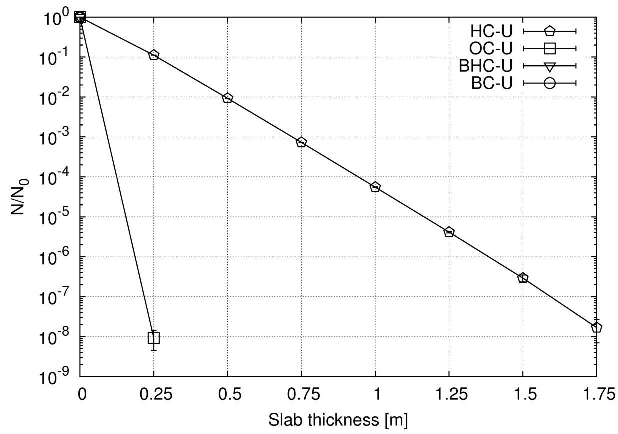

Fig. 18. Monoenergetic neutrons attenuation in Ulysse ordinary concrete

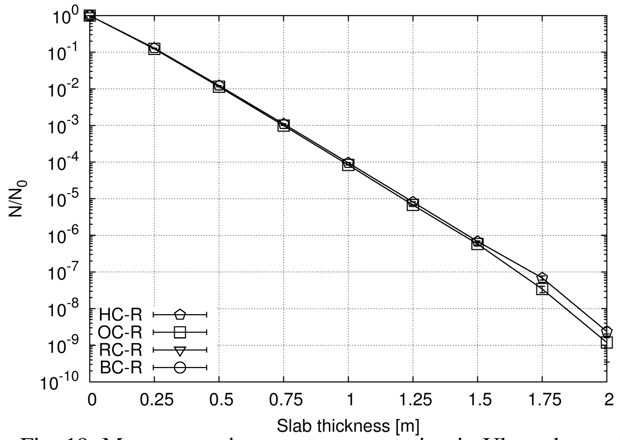

Fig. 19. Monoenergetic neutrons attenuation in Ulysse borated concrete

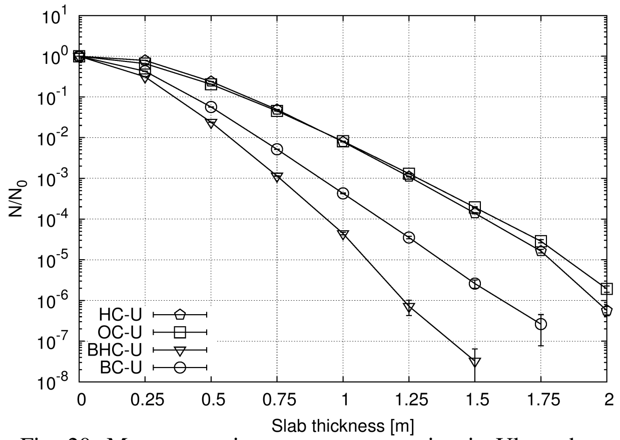

Fig. 20. Monoenergetic neutrons attenuation in Ulysse heavy concrete

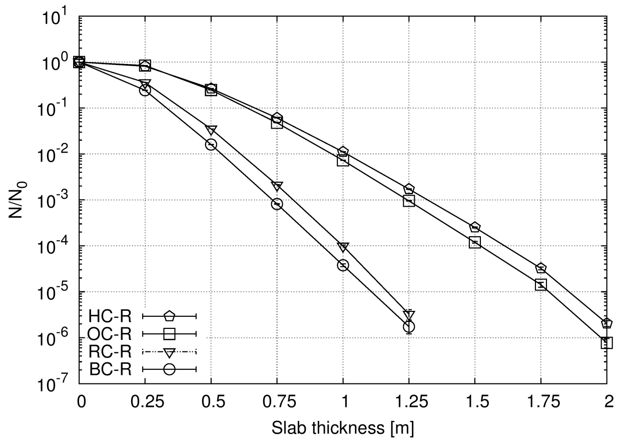

Fig. 21. Monoenergetic neutrons attenuation in Ulysse borated heavy concrete

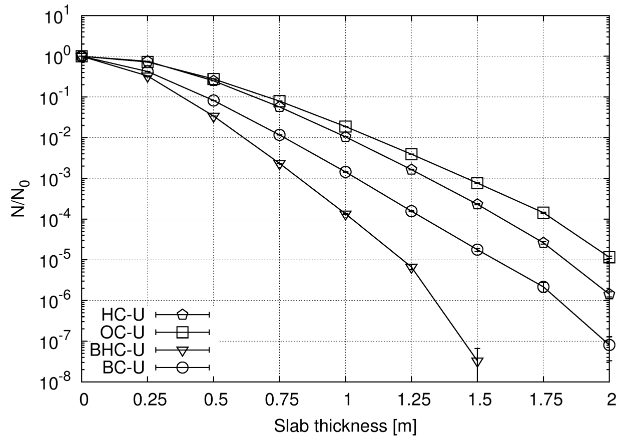

Fig. 22. Monoenergetic neutrons attenuation in RUS ordinary concrete

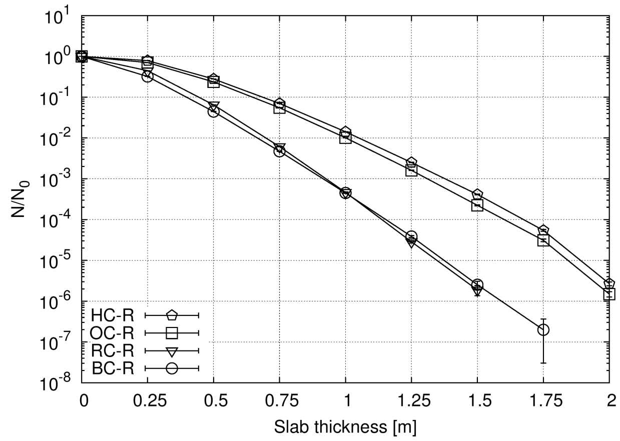

Fig. 23. Monoenergetic neutrons attenuation in RUS borated concrete

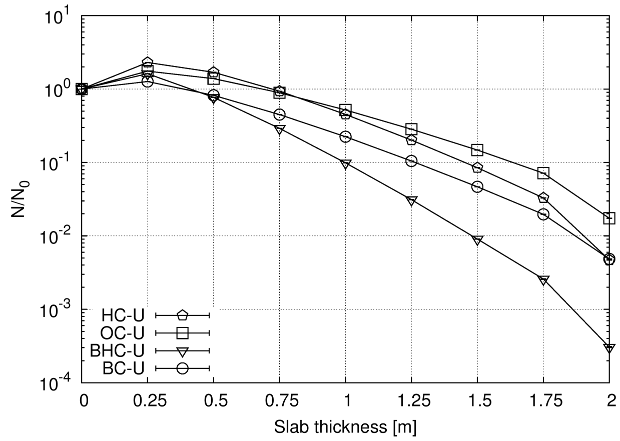

Fig. 24. Monoenergetic neutrons attenuation in RUS heavy concrete

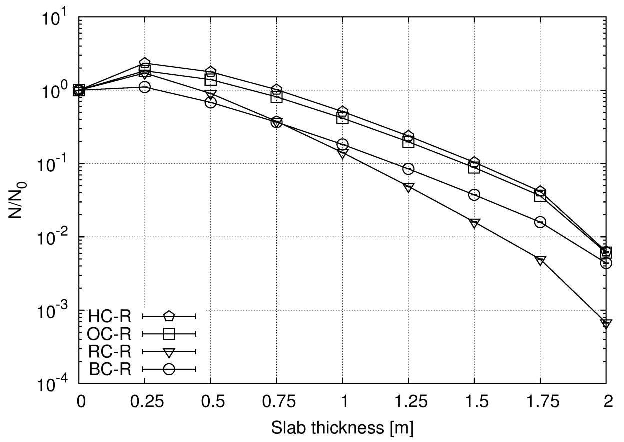

Fig. 25. Monoenergetic neutrons attenuation in RUS reinforced concrete

6. Conclusions

The aim of this paper is to show the usefulness of a computer-aided design of neutron shielding concretes. The shielding efficiency against gamma radiation strongly depends on the density of the material but neutron radiation shielding is a much more complex phenomenon. Neutron shielding should be solved taking into consideration two aspects: slowing down neutrons, which is best done with light nuclides (hydrogen) and neutron absorption with nuclides with a high cross section for the neutron absorption like boron. That is why the results of neutron attenuation for borated concrete in comparison to heavy concrete is of one or two orders of magnitude. When a new shielding material is developed not only the shielding efficiency should be taken into account, but an activation of material as well. The last depends on the chain decay of many nuclides and it develops in time, in a complex manner. Here the advantage of borated concretes is removed as they have been the most activated and their activation decrease in time has been the slowest. From this point of view the best solution for neutron shielding concrete is to reinforce one because its shielding properties (attenuation) are very good and high activation at the beginning becomes very small just after 50 years of cooling. It leads to a conclusion that time of operation and time of dismantle should be carefully chosen in the shielding concrete design as well.

This study of neutron shielding is more important when not only thermal neutrons should be attenuated. The worst concrete for this purpose seems to be the barite heavy concrete commonly used against gamma and other radiation. It is full of high mass elements with very small cross sections for neutron capture. The presented results are the second step of the research program. The first were the simulations on shielding of theoretical composition of concretes in a theoretical neutron flux [24]. In this paper there have been simulated shielding and activation of real concretes from nuclear facilities in a theoretical neutron flux. Now the authors are looking for a support in validation of these simulations on real concretes in a real neutron flux.

Acknowledgements. This work was prepared during the program of the Polish university staff training for nuclear engineering in France in 2011. We would like to express our deep gratitude to organizers of this stage: the Polish Ministry of Economy and l' Agence France Nucléaire International.

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