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ATTENUATION OF NEUTRON AND GAMMA RADIATION BY A COMPOSITE MATERIAL BASED ON MODIFIED TITANIUM HYDRIDE WITH A VARIED BORON CONTENT

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The investigations on estimating the attenuation of capture gamma radiation by a composite neutron-shielding material based on modified titanium hydride and Portland cement with a varied amount of boron carbide are performed. The results of calculations demonstrate that an introduction of boron into this material enables significantly decreasing the thermal neutron flux density and hence the levels of capture gamma radiation. In particular, after introducing 1- 5 wt.% boron carbide into the material, the thermal neutron flux density on a 10 cm-thick layer is reduced by 11 to 176 factors, and the capture gamma dose rate – from 4 to 9 times, respectively. The difference in the degree of reduction in these functionals is attributed to the presence of capture gamma radiation in the epithermal region of the neutron spectrum.

Keywords: titanium hydride, Portland cement, boron carbide, neutron radiation, gamma radiation, fast neutrons, thermal neutrons, functionals.

INTRODUCTION

One of the promising neutron-shielding materials is titanium hydride having $9.12 \cdot 10^{22}$ hydrogen atoms per 1 cm^3 of the material. In contrast to filled polymers, titanium hydride exhibits higher service temperatures and acceptable neutron flux density. The major competitor of titanium hydride in the world market is an exclusive material RX-277 (USA), which is traditionally used in nuclear power engineering for shielding systems and in storage containers for nuclear waste. However, RX-277 possesses lower neutron-shielding properties than titanium hydride (whose neutron radiation dose rate behind a shield of the same thickness is by a factor 2.7 higher), which demonstrates the advantage of the latter material [1].

The use of titanium hydride briquettes, powder, or chips implies certain restrictions, such as low operating temperature (up to 200°C), in view of the fire- and explosion danger of the shielding structure due to hydrogen release. To overcome this limitation, it is popular to use titanium hydride pellets (shots) manufactured by the method of centrifugal atomization of titanium followed by hydrogenation. Modification of the shots with borate compounds allows increasing the thermal stability and rules out thermal diffusion of hydrogen up to the temperature 500–550°C [2–4].

It is critical to reduce the output gamma radiation formed during the reactor operation in the bioshield materials and structures as a result of capture of slow neutrons, which is especially urgent for vehicle-borne small dimensional nuclear power plants. One of the ways to reduce the capture gamma radiation output is to introduce an additive into the material, which would contain an isotope with a large neutron-absorption cross-section in the thermal and epithermal spectral regions. In nuclear industry, mostly boron and its compounds are used for this purpose, which also exhibit low outputs of their intrinsic gamma radiation. An introduction of boron results in a decrease in the fraction of thermal neutrons in the spectrum, which in turn reduces the intensity and the dose rate of capture gamma radiation due to

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TABLE 1. Atomic Concentration of Elements in the Experimental Composite with Variable Content of B_4C

B_4C, wt.%	Atomic concentration·10^{-20}, 1/cm^3										
	Ti	Si	Al	Fe	Ca	Mg	H	O	C	B10	B11
0	298	19.9	8.99	2.43	45.4	1.91	474	104.0	0.874	0.0	0.0
1	298	19.1	8.83	2.35	43.4	1.83	474	99.80	4.360	2.63	11.3
2	298	18.2	8.68	2.26	41.3	1.74	474	95.30	7.750	5.25	22.6
3	298	17.3	8.52	2.18	39.2	1.65	474	90.80	11.30	7.88	34.0
4	298	16.4	8.36	2.09	37.2	1.57	474	86.20	14.80	1.05	45.3
5	298	15.5	8.21	2.00	35.1	1.48	474	81.70	18.30	1.31	56.6

reactions of the (n, γ) type, as well as the activation processes in the thermal spectral region, and hence reduces the induced activity of the material [5].

In this study we present the results of model calculations of attenuation of the reactor emissions in the material based on modified titanium hydride and a Portland cement binder with a varied content of boron carbide in it, in order to estimate the degree of reduction of the thermal neutron flux density and the level of capture gamma radiation. An addition of boron, having a large cross section of neutron absorption in the thermal and epithermal spectral regions, is one of the means for improving its shielding characteristics.

MATERIALS AND EXPERIMENTAL PROCEDURE

The bioshield material was manufactured in accordance with the proposed technology by the method of compaction of a composite mixture of the modified titanium hydride and Portland cement pellets, followed by solidification and drying to achieve the composite bulk density 3.2 g/cm^3.

In order to perform variant calculations and obtain spatial-energy distributions of the thermal neutron flux densities and gamma radiation in the bioshield material, we used an arrangement consisting of a nuclear reactor core, a core reflector, structural materials, and a 1 m-thick layer of the material under study. The model radiation source taken for this purpose was an MBIR multipurpose fast-neutron reactor providing the thermal neutron flux density $2 \cdot 10^{15}$ cm^{-2} s^{-1}.

The calculations were performed using an ANISN code [6], providing a solution to the one-dimensional transport equation by the method of discrete ordinates taking into account scattering anisotropy. The spectrum of neutrons was computed for a 22-group partition of the energy interval. The gamma radiation spectrum had an 18-group partition [7–9].

The composition of the experimental material was simulated using the content of boron carbide (B_4C) from 0 to 5.0 wt.% and the amount of bound water removed from the material during its drying in accordance with the results of the experimental investigations. It should be noted that the introduction of carbide was simulated using the reduced content of Portland cement in the composite under study.

The atomic concentrations of the elements in the experimental material, used in the ANINS calculations, are presented in Table 1.

A special feature of the calculation for this problem is the need to single out a source of capture gamma radiation proper from all potential generators affecting the flux of gamma rays in the material under study, ruling out the leakage of intrinsic γ-radiation from the core and the structure materials. To achieve this, we used a certain arrangement where an additional water tank (15 cm) and a lead shield (15 cm) were placed between the pressure vessel of the reactor and the material under study. The lead shield before the material serves to minimize the leakage of gamma radiation from the core, the reactor internals, and the reactor vessel, which makes it possible to distinguish the source of intrinsic capture gamma radiation in the material under study from all its potential generators (materials of the core, vessel, and internals).

TABLE 2. Values of Coefficient K_{nt} Characterizing the Neutron Flux Density Reduction in the Shield Material under Study after Introduction of B_4C into its Composition

Content of B_4C, wt.%	Layer thickness, cm					
	5	10	20	30	40	50
1	12.5	11.0	10.8	10.7	10.6	10.6
2	34.2	30.6	29.8	29.5	29.4	29.4
3	69.3	62.6	61.0	60.6	60.4	61.7
4	120.0	109.5	107.3	107.3	107.7	110.4
5	191.4	176.3	173.8	174.6	176.1	180.6

TABLE 3. Values of Coefficient K_γ Characterizing the γ-Dose Rate Reduction in the Shield Material under Study after Introduction of B_4C into its Composition

Content of B_4C, wt.%	Layer thickness, cm					
	5	10	20	30	40	50
1	4.2	4.3	4.6	4.7	4.7	4.6
2	5.9	6.3	6.9	7.1	7.1	7.1
3	7.0	7.5	8.6	8.9	8.9	8.8
4	7.8	8.5	9.8	10.3	10.3	10.1
5	8.3	9.2	10.8	11.3	11.3	11.2

RESULTS AND DISCUSSION

Tables 2 and 3 present the coefficients characterizing the reduction in the thermal neutron flux density (K_{nt}) and dose rate of γ-radiation (K_γ), resulting from the introduction of the boron carbide addition into the material based on the modified titanium hydride and Portland cement pellets. The K_{nt} and K_γ coefficients are determined as the ratio of the functionals in the material without and with boron carbide.

From the data presented in Tables 2 and 3 it is evident that the boron carbide addition significantly reduces the capture gamma radiation dose rate (P_γ) and even more so reduces the thermal neutron flux density. This difference could be attributed to the contribution into P_γ from the neutron capture in the epithermal energy region, where the boron carbide addition does not play any appreciable role. Nevertheless, an introduction of up to 5 wt.% of boron carbide gives rise to a decrease in K_γ from a factor of 8 to 11 depending on the layer thickness.

A certain decrease in K_{nt} with increasing the layer thickness of the shield material under study results from an increased thermal energy relaxation length for the flux of thermal neutrons due to the production of capture gamma radiation.

For the sake of illustration, Figs. 1 and 2 present exponential distributions of thermal neutron flux density and dose rate of the intrinsic capture gamma radiation in the material compositions with different boron concentrations.

The pattern of gamma dose rate distribution over the shield layer thickness and its value behind the shield determine the radiation leaking to the front wall and the capture gamma radiation forming in the initial layer of a few centimeters in thickness (note that the former component is lower than the latter one). Therefore, the value of P_γ behind the material is primarily determined by the source of gamma radiation located either in its initial layer or in front of the layer; the generation of capture gamma radiation in the rest of the material does not make any significant contribution. As a result, the relaxation length of the gamma radiation dose rate in the shield material under study does not virtually depend on the content of hydrogen in it.

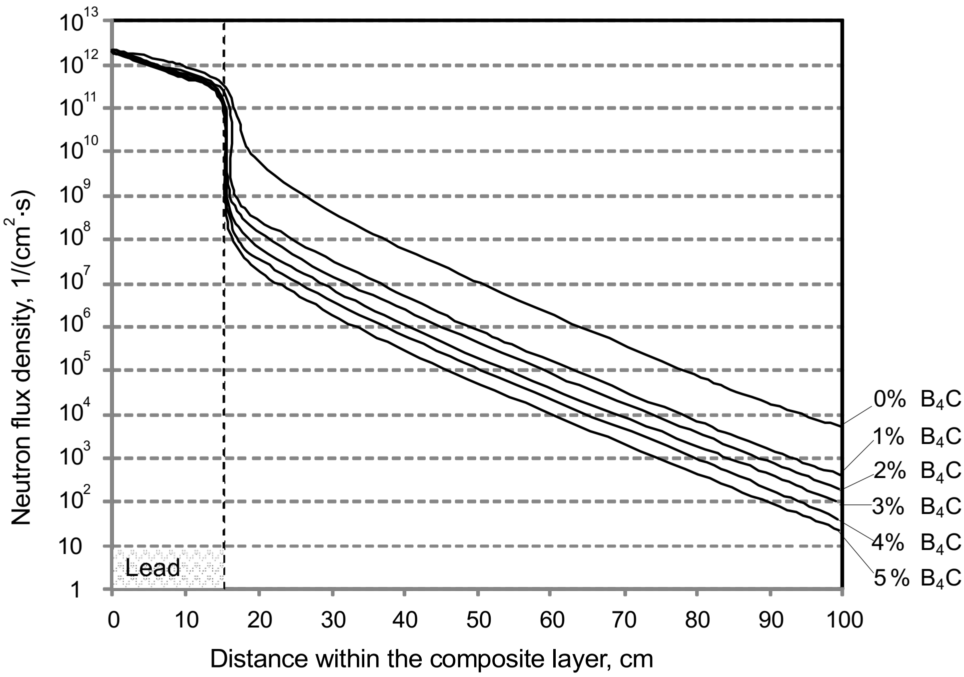

Fig. 1. Flux density distribution in the shield material based on the modified titanium hydride pellets with variable content of boron carbide (wt.%).

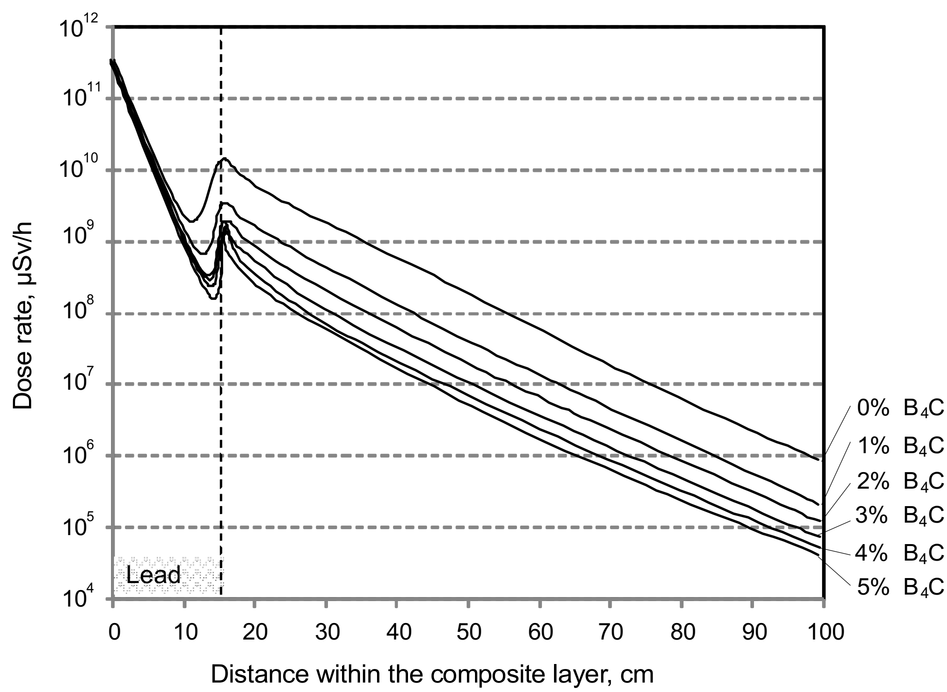

Fig. 2. Distribution of capture gamma radiation in the shield material based on modified titanium hydride pellets with differing additions of boron carbide (wt.%).

To conclude, an addition of boron, having a large neutron absorption cross section in the thermal and epithermal spectral regions, into the shield material gives rise to a decrease in the thermal neutron flux density and capture gamma radiation dose rate.

CONCLUSIONS

The calculated data on reducing the emissions from the reactor shield material based on modified titanium hydride and Portland cement with a boron carbide addition have been presented. The result of calculations demonstrate that an introduction of boron into the material allows decreasing the thermal neutron flux density and hence the level of capture gamma radiation. In particular, an addition of boron in the concentration from 1 to 5 wt.% decreases the flux density within a 10 cm-thick material layer by 11 to 176 factors and the capture gamma radiation dose rate – by 4 to 9 factors. The difference in the level of reduction of these functional is due to the presence of capture gamma radiation in the epithermal spectral region.

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