Neutron and gamma radiation shielding properties of high-temperature-resistant heavy concretes including chromite and wolframite

B Aygün
Department of Electronics and Automation, Agri Ibrahim Cecen University, Agri, Turkey

ABSTRACT
High performed new heavy concrete samples were designed and produced that absorption parameters were determined for gamma and neutron radiation by using Monte Carlo Simulation program GEANT4 code. In the sample production, many different materials were used such as: chromite (FeCr₂O₄), wolframite ((20Fe,80Mn) WO₄), hematite (Fe₂O₃), titanium oxide (TiO₂), aluminum oxide (Al₂O₃), limonite (FeO(OH)ₙH₂O), barite (BaSO₄), materials. Furthermore, calcium aluminate cement (CAC) was utilized for high temperature resistance. In the current study, five different new heavy concrete samples were produced then physical and chemical strength of them tested. High-temperature-resistant tests were made at 1000°C and good resistance against high temperature was observed. Neutron equivalent dose measurements were made for by using 4.5 MeV energy ⁴¹⁰Am-Be fast neutron source. Results compared with paraffin and conventional concrete. It was found that the new heavy-weight concretes had the better absorption capacity than paraffin and conventional concrete. Gamma radiation absorption measurements also were carried out at the energies of 160, 276, 302, 356, and 383 keV by using ¹³³Ba point radiation source. It has been suggested that the new produced concretes can be used for radiation safety in the nuclear applications.

1. Introduction
Radiation is often used in applications such as in energy production, in medicine diagnosis and treatment, in material research and investigation. In addition, it is also used in such areas as agriculture, archeology (in carbon determination), space exploration, military, geology, and many others (U.S. NRC, 2010). Radiation leaks may occur during these applications (Lamarsh, & Baratta, 2001); therefore, it must be properly shielded. In radiation shielding works, conventional materials such as concrete, steel, alloy, ceramic, glass, and polymers are widely used (Aygün et al., 2019; Kumar, Sayyed, Dong, & Xue, 2018; Sayyed, Akman, Kumar, & Kaçal, 2018). In these studies, concrete is among the most widely used materials (Li et al., 2017). Concrete is a composite material which glued in such a way that aggregate particles (sand, gravel, stone, and filler) with cement or a binder. Traditional concrete is not as effective in shielding radiation, but it is a very common used building material. The traditional concrete shielding characteristic may vary and is dependent on the chemical composition of the concrete. New types of concrete samples have been developed by different the aggregated used for preparing concrete, depending on the available natural and artificial materials (Mukhtar, Shamsad, Al-Dulaijan, Mohammed, & Akhtar, 2019; Chen, 1998). Heavy concrete is the most common material used in radiation shielding. Heavy concrete is obtained by adding high-density aggregates into normal concrete. Normal-weight concrete density varied between 2200 and 2450 kg/m³ while heavy concrete’s density is ranging from about 2900 and 6000 kg/m³ (Nawy, 1997). Some natural minerals such as hematite, magnetite, limonite, serpentine, siderite and barite can be used as aggregates in heavy concrete production. In literature, numerous experimental and theoretical researches have been conducted to develop new heavy concrete. Different minerals like siderite, limonite were used to produce heavy concrete in order to provide gamma radiation shielding. It was reported that the gamma radiation absorption capacity of heavy concretes is high (Basyigit et al., 2011). Boron-containing multi-layered new heavy concretes were produced and radiation shielding properties were determined. It is reported that these concretes are very high in 14 MeV neutron absorption capacity (Sato, Maegawa, & Moshimatsu, 2011). In a different study, some metal oxides such as Al₂O₃, As₂O₃, BaO, CaSO₄, CdO, Cr₂O₃, CuO, Fe₂O₃, K₂O, MgO, MnO, Na₂O, NiO, P₂O₅, PbO₄, SrO, TiO₂ was used in the heavy concrete production, and it was stated that the use of these new heavy concretes in nuclear reactors is appropriate (Abdo, 2002; Erdem, Baykara, Doğru, & Kulüözü Türk, 2010; Mortazavi, Mosleh-Shirazi, & Baradaran Ghafrarokhi et al., 2010). Selbort et al.produced heavy concretes by using, such as calcium (Ca), strontrium (Sr), barium (Ba), radium (Ra)
magnesium (Mg) elements. They determined these heavy concretes can be used to shield gamma and neutron radiation in nuclear reactors (Selberg et al., 2005). In the present study of tungsten oxide (WO₃) gamma radiation mass attenuation coefficient in the concrete, the effect on the coefficient was investigated. Appropriate geometry found by using MCNPX and XCom simulation programs. It is found that shielding properties when nanoparticle WO₃ doped in concrete more than microparticle WO₃ (Tekin, Singh, & Manici, 2017). In another study, high-density concrete (ρ = 4.71 g/cm³) was made by using steel balls and in aggregate the debris of the demolished concrete buildings in the earthquake region in Fukushima. Good shield properties were determined this of heavy concrete and it is shown that can be used in storage radioactive waste (Sanjay, Yusuke, Kimura, Fujikura, & Araki, 2018). Heavy concrete was made using lead-zinc slag waste instead of sand which can be used gamma radiation shielding. Shielding and strength properties were investigated of this concrete and compared with conventional concrete. It is reported that lead–zinc slag waste concretes better radiation shielding and strength characteristic than conventional concretes (Mohamed, 2017). Medical cyclotron is a system designed for radiopharmaceutical production, which high-level radiation emit. Shielding wall thickness was calculated by using Monte Carlo simulation when cyclotron system used to operate that may occur radiation. Consequently, for shielding, radiation at 200-cm-thickness concrete wall need was determined (Jang, Kim, & Kim, 2017). Some mining wastes suitable for heavy concrete production. For instance, Gallala et al. have produced new heavy concrete by using barite-fluorspar mine waste (BFMW) aggregates and investigated their gamma radiation shielding, mechanical strength properties. The results clearly showed when ratio 25% BFMW added to concretes has better gamma radiation shielding and compressive strength properties than conventional concrete (Gallala et al., 2017). Tekin et al., using MCNPX code, demonstrated that high strength concrete containing nanoparticles of WO₃ and Bi₂O₃ had enhanced shielding capacity for gamma radiation (Tekin, Sayyed, & Issa, 2018). Five different concrete types were made using magnetite aggregates and 0%, 2%, 4%, 6%, and 8% of titanium dioxide (TiO₂) nanoparticles for nucleon power plant shielding material. Some of the protecting parameters such as MAC (mass attenuation coefficients) HVL (half-value layer), TVL (tenth value layer), and linear attenuation coefficients (LAC) were determined for 662, 1173, and 1332 keV energy of gamma ray used. It is reported, the significant effect on radiation shielding properties occurred within 8% of TiO₂ nanoparticles (Iman et al., 2019). Some natural minerals can be using heavy concrete in production. Different concrete types which including natural perlite mineral and B4C have been experimentally investigated and gamma radiation shielding parameters have been determined (Agar et al., 2019).

In this study, new concrete samples were designed and produced using Monte Carlo simulation program Geant4 code. The production of heavy concrete for radiation shield was made based on the concrete production process such as mixture proportion, ratio of water to cement, cement hydration. Furthermore, new concrete candidates with good radiation shielding ability at high temperature have been produced and it has been shown that raw materials such as chromite, wolframite can be used in production.

2. Material and method

2.1. Monte Carlo simulation codes Geant4

In Monte Carlo simulation program, the Geant4 code is used to determine the interactions between radiation and materials. In addition, it can be used to predict nuclear events that may occur at the point of radiation and detector interaction. Geant4 software is the most developed, for analyses biological effects of radiation-induced and their modification applications shielding. Also, Monte Carlo program Geant4 to simulate can be used to predict the transport, accumulation of incident particles through the walls of a nuclear power plant (Agostinelli et al., 2003). It is used in applications in nuclear physics, particle accelerator designing, space investigation, and medical physics. Detailed information can be found at www.Geant4.org.

2.2. Sample preparation

New heavy concrete samples were produced by using different natural aggregates such as chrome ore (FeCr₂O₄), wolframite (Fe,Mn)WO₄, hematite (Fe₂O₃), limonite (FeO (OH) nH₂O), barite (BaSO₄). Nickel oxide (NiO) was used to fill the pores that could form in the concrete. The chromium ore (FeCr₂O₄) mineral has a density of average 4.79 g/cm³ and it melts in temperature 1650–1660°C (Jay, Meegoda, Zhengbo, & Kamolpornwijit, 2007). The chrome ore sample was taken from the Kayseri city Yahyalı district chrome mine. This chrome ore contains such minerals 53.19% Cr₂O₃, 16.80% MgO, 11.51%Al₂O₃, 15.11%Fe, 2.72%SiO₂, 0.007%S, and 0.005% Fe according to Eti (Chromium Ferrochrome Foundation). Wolframite is a mineral with a density of 7.1–7.5, average 7.3 g/cm³ and 11.70% MnO, 16.85% FeO, 71.46% WO₃ including (Tolun, 1951). This ore was obtained from an Uludağ tungsten mine, which is located in the province of Bursa and is approximately 2200–2300 m high from the sea. According to the pioneering simulation work, both gamma and neutron radiation absorption cross-sectional values were determined higher in chromite and wolframite minerals.
Furthermore, these minerals have both refractory properties and high mechanical strength and plenty of reserves. Therefore, these minerals were used in the production of heavy concrete. Hematite, titanium oxide, aluminum oxide, limonite, siderite, barite, materials are always used materials for the production of heavy concrete, but for that, the chromite and wolframite minerals are not very commonly used. The usage of natural chromite and wolframite minerals provided will be with this work in the nuclear industry. Chromium oxide (Cr₂O₃) was used to fill capillary cavities that may form in concretes. When concrete components were selected, the high macroscopic cross-sectional values were taken into account. Before the production of new heavy concrete, sample materials and percent mass component ratio were determined by using Monte Carlo simulation program Geant4 code. The water absorption rates of the aggregates of the sample to be used in the production were calculated. Prior to operation of aggregates dry weight (W₁) was determined. Then, aggregates were kept in water 24 h, dried in a 110°C and weighed (W₂) found. Water absorption ratio (%) calculated by using a formula,

\[
\text{water absorption ratio (\%)} = \frac{(W₂ - W₁)}{W₁} \times 100
\]

The results of the water absorption capacity and specific gravity of the aggregates are shown that Table 1.

The determined water absorption ratio and specific density of the aggregates are given in Table 1. It has been found that the water absorption ratios of the aggregates and compared with American Society for Testing and Materials (ASTM) standards, obtained results were consistent with these standards (Archor, Allen, Hughes, Quinion, & Thorpe, 1988). According to ASTM standards, water absorption on aggregates should not exceed 3%

New heavy concrete samples were produced by using a conventional production technique. Aggregates, nickel oxide, and cement were mixed for 10 min in dry condition, and then water was added at the specific ratios and a homogeneous mortar mixture was realized. After prepared mortar was placed in molds and dried at room temperature. The samples were kept in lime water for 28 days and determined the water, abrasion resistance according to ASTM C 31–39 standards. Flame and high-temperature resistance of samples were tested for 6 h by using a bunsen burner, respectively, at 573.15 K, 773.15 K, 1023.15 K, 1273.15 K. Compressive strength test was carried out to find the fracture strengths of the samples under load. The samples were placed vertically on the platform of 25-ton capacity hydraulic press machine which has a 600 MPa capacity. The force applied constantly and uniformly also the applied force was decided until the samples began to break. Compressive strength was identified according to the following formula.

\[
f_c = \left( \frac{F}{A_C} \right) \text{(N/mm}^2)\]

f_c: Concrete pressure resistance (N/mm²) F: The value at which concrete begins to break (N) A_C: The cross-sectional area through which the sample is loaded. (mm²). It was determined that all new heavy concrete samples exhibited strength up to 30 MPa (4351.131psi).

Chemical composition and densities of samples are shown in Table 2. Also, produced samples can be seen in Figure 1.

The fast neutron (4.5 MeV) radiation absorption experiments were carried out by using Canberra brand portatif NP series BF₃ gas neutron detector and Am/Be fast neutron source which emits neutrons in the 2–11 MeV range, but has an average energy of 4.5 MeV. Samples were placed between the source and detector and then exposed to radiation. The samples absorption dose rates were calculated by subtracting the dose emitted by the source and the dose absorbed by the detector. The experimental design is shown in Figure 2.

As shown in Figure 3, an experimental mechanism established and for the gamma radiation absorption measurements, ¹³⁷Ba 10 mCi point source and DSG planar HPGe (high purity germanium) detector were used. Measurement time of the samples was taken at 1200 s (live time) and standard deviation values were calculated for each sample measured for three times.

3. Results and discussion

3.1. Neutron shielding parameters

3.1.1. Total macroscopic cross section

Fast neutrons are difficult to capture since its interaction is possibly being low with the material nucleus. Therefore, for shielding, fast neutrons must be slowed down by colliding with shield material before being captured. This collision may cause elastic and inelastic scattering. Elastic scattering, inelastic scattering, and capture may be expressed by the total macroscopic cross section (\(\Sigma = \Sigma\text{scat} + \Sigma\text{capture} + \Sigma\text{fission} + \ldots \text{cm}^{-1}\)) (Lilley, 2001). The macroscopic cross section shows the possibility of interaction of shield material with neutrons, how

| Aggregate     | Water absorption capacity (mass %) | Specific density (g/cm³) |
|---------------|-----------------------------------|--------------------------|
| River sand (2–4 mm) | 3.56                              | 2.56                     |
| Chromite (2–4 mm)   | 1.27                              | 3.79                     |
| Barite (0–2 mm)     | 0.58                              | 4.04                     |
| Wolframite (2–4 mm) | 0.41                              | 2.06                     |
| Hematite (2–4 mm)   | 1.24                              | 2.42                     |
| Limonite (2–4 mm)   | 1.79                              | 4.03                     |
big is this probability the higher the neutron absorption capacity of the material (Ott et al., 1989–1985). As shown in Table 3, theoretical Total Macroscopic Cross Sections values were determined for new heavy concrete and the reference sample selected conventional concrete and paraffin wax that they are widely used in neutron shielding work.

Obtained results were compared with conventional concrete and paraffin; it is found that total macroscopic cross-sectional values of new heavy concrete (NHC1, NHC2, NHC3, NHC4, NHC5) samples are higher than the reference samples. It has been seen that, in particular, the NHC1 sample had a high total macroscopic cross section. This comparison is shown in Figure 4. From this figure, it is observed that for the new heavy concrete samples, have higher values of total macroscopic cross sections than paraffin and conventional concrete.

### 3.1.2. Mean free path

The average distance that a neutron can receive without any interaction in the material or average distance between the two interactions is expressed by the mean free path \( \lambda \). Interaction probability is very low of the neutron at this distance. There is a relationship between the mean free path and the total macroscopic cross-section of a neutron and this can be shown as

\[
\lambda = \frac{1}{\sum}
\]

That is, if a material have a large total cross section, the average free path is small, indicating that the material has a high neutron absorption ability (Lamarsh, 1983, 2001). The obtained values of mean free path values for reference (paraffin, conventional concrete) and new heavy concrete samples are shown in Table 3. These values show that the minimum value of the mean free path is observed in the new heavy concrete samples. According to the results, we report that the lower mean free path values of the new heavy concrete samples the better radiation shielding materials in comparison to paraffin and conventional concrete shielding concretes.
3.1.3. Transmission

A total of 100,000 fast neutron with 4.5 MeV energy sent on each sample and passing neutron numbers were determined by using the Monte Carlo Simulation technique. As shown in Table 3, obtained results compared with paraffin and conventional concrete, it has been seen that passing transmission neutron number low from new heavy concrete samples than reference samples. The decrease in the number of neutrons passing through these new concretes depends on the reduction in the free path because of the chromite and wolframite in the concrete structure. Internal structures of new heavy concrete were strengthened by the atoms of these elements, which prevents the formation of loose-packaged structures.

Low passing neutron number is an indication for the high shielding ability of new heavy concrete samples.

3.1.4. Equivalent dose rate measurements

Equivalent dose measurements of new heavy concrete samples were carried out by using 4.5 MeV $^{241}$Am-Be fast neutron source and BF$_3$ gaseous detector. By measuring the dose from the source ($D_0$), by calculating passing dose from the sample ($D_D$) and by taking the difference between the two (i.e. $D_0-D_D = D_S$), samples absorbed dose ($D_S$) was determined. The results were compared paraffin and conventional concrete, as shown in Table 4 and in Figure 5.

As shown in Table 4 and in Figure 5 this could be concluded that the dose of 1.485 ($\mu$Sv/h) emitted from the source was absorbed by paraffin (by 38.44%), by normal weight concrete (by 27.38%), by NHC1 (by 57.99%), by NHC2 (by 51.25%), by NHC3 (by 54.12%), by NHC4 (by 54.12%) and by NHC5 (by 50.12%). These results clearly show that the dose rates absorbed by new heavy concretes are higher compared to conventional concrete and paraffin. All samples have perfect shielding performance, but NHC1 is much better than others.

### Table 3. Total macroscopic cross sections, Mean Free Path, Transportation values of samples 3 cm thick.

| Sample code | Total macroscopic Cross section (cm$^{-1}$) | Mean Free Path (mm) | Transmission Number |
|-------------|---------------------------------|------------------|-------------------|
| CC          | 0.508              | 1.377 ± 8.5909   | 60,166            |
| P           | 0.6145             | 1.356 ± 8.5825   | 54,115            |
| NHC1        | 1.1219             | 1.2248 ± 8.4033  | 32,563            |
| NHC2        | 0.8945             | 1.2738 ± 8.4876  | 40,879            |
| NHC3        | 0.9777             | 1.2542 ± 8.4553  | 37,616            |
| NHC4        | 1.0092             | 1.2569 ± 8.4406  | 36,451            |
| NHC5        | 0.850833           | 1.2917 ± 8.5012  | 42,706            |

CC: Conventional concrete, P: Paraffin, NHC: New heavy concrete

### Table 4. Experiments equivalent neutron dose rates.

| Sample code | Absorbed equivalent dose rates ($\mu$Sv/h) by the detector ($D_D$) | Absorbed equivalent dose rates ($\mu$Sv/h) by the samples ($D_S$) | Absorbed dose percentage of samples (%) |
|-------------|---------------------------------------------------------------|---------------------------------------------------------------|--------------------------------------|
| Background  | 1.485                                                         | -                                                             | -                                    |
| Paraffin    | 0.811                                                         | 0.574                                                         | 38.44%                              |
| CC          | 0.978                                                         | 0.407                                                         | 27.38%                              |
| NHC1        | 0.631                                                         | 0.854                                                         | 57.99%                              |
| NHC2        | 0.722                                                         | 0.763                                                         | 51.25%                              |
| NHC3        | 0.683                                                         | 0.802                                                         | 54.12%                              |
| NHC4        | 0.674                                                         | 0.811                                                         | 54.61%                              |
| NHC5        | 0.739                                                         | 0.746                                                         | 50.23%                              |

Figure 3. Gamma radiation measurement system.

Figure 4. Theoric 4.5 MeV Neutron Total Macroscopic Cross Section.

Figure 5. Experimental equivalent dose measurement ($\mu$Sv/h) of the samples.
is due to the fact that NHC 1 sample contains high ratio hematite, equal ratio chromite, and wolframite mineral.

### 3.1.5. Neutron shielding calculation

Thicknesses of new heavy concretes were calculated that can stop 4.5 MeV energy fast neutrons. Thus, optimum concrete thickness was determined and results compared with paraffin, conventional concrete. Equation 3 (Schaefee) used in calculations.

\[ D = B D_0 e^{-\Sigma TX} \]

D is the dose rate absorbed by shield material (µSv/h), \( D_0 \) is the dose rate without shield material (µSv/h), \( \Sigma T \) is the total macroscopic cross section (cm\(^{-1}\)), T is the neutron cross section \( \Sigma \) fission + \( \Sigma \) escaping + \( \Sigma \) captured + \( \Sigma \) ... + X is the thickness of shield material (cm), B is a buildup factor usually assumed to be 5 for Am-Be fast neutron source. As expressed in equation 3, the absorbed dose rate change depending on the total macroscopic cross section and the thickness of the material. For this reason, macroscopic cross-sectional values were calculated for different thicknesses of every sample, and then used equation 3 and results are shown in Table 5.

For each shield samples, passing dose rate has been calculated for up to thicknesses from 3 cm to 30 cm. The calculated values indicated that the absorbing dose rates of the shielding materials depend on the thickness of the samples shown in Table 5. Increasing thickness of the samples shows a high absorption dose, which means that the probability of interaction with fast neutrons increases as the samples became thicker. It was also observed that at 24 cm thickness no radiation dose passed from NHC1-3-4 samples and at 27 cm thickness doses not passed from NHC2-5 samples. But, according to calculation the conventional concrete at the thickness of 43 cm and paraffin at the thickness of 36 cm completely absorbed 1.485 (µSv/h) dose. These results are evidence that they are perfect samples for 4.5 MeV energy fast neutron radiation shield. When the concrete is exposed to heat, it is continually heated up and the accumulated temperature flows from the inside to the outside and expansion starts. If concrete aggregates are not durable, these expansions, after a while tensile stresses and tensile strength are exceeded, cracking may be occurred. So, the concretes applied in nuclear reactors not only must have high ability of radiation dose but also high temperature resistant. Taking into account this situation, high-heat-resistant chromite, wolframite aggregates, and aluninate cement were used in the production of samples. The samples were observed as possible effects directly by holding the heat at 300-600-900°C. After the heat treatment, again radiation shielding experiments were carried out and it was determined that 0.001% of the radiation passed. The results also indicate that there were invisible capillary cracks in the concretes, but these are not very important since those cracks are nano-sized. Furthermore, when the temperature reached at 500°C, the ceramic bond between the aggregate and the cement was formed, so that the concrete was found to be even stronger.

### 3.1.6. Gamma radiation shielding properties

Experimental mass attenuation coefficients (MACs) of the new produced heavy concretes measured by using \(^{133}\text{Ba}\) radioactive source and HPGe detector. The experimental results were calculated in six different gamma energies (160, 276, 302, 356 and 383 keV) using well-known Beer–Lambert law (Figure 6(a)). Radiation protection efficiencies

\[ (\text{RPE}(\%) = (1 - I/_{I_0}) \times 100) \]

were also estimated for the given concretes (Figure 5(b)). It is clearly seen from Figure 6(a and b), the experimental results are highly dependent on the chemical concentrations of the sample and the incident photon energies. It is seen from Figure 6(a); the MAC values are decreasing with the increasing photon energies due to high penetration properties of these photons. Also from this figure, it was concluded that increasing chrome ore (FeCr\(_2\)O\(_3\)) and wolframite [(Fe, Mn) WO\(_4\)] ratios in the sample significantly increase the absorption values of produced concretes. The best MAC values were seen in the sample of NHC5 in all measured energies. For this reason, the sample of NHC5 has to have the highest effective atomic number and electron density among the produced concretes. Figure 6(b) is supporting to the MACs results because the maximum RPE values are obtained in the sample of NHC5 in all energies.

### 4. Conclusion

In this study, five new heavy concrete samples containing chromite, wolframite, hematite, limonite, barite, nickel oxide aggregates were produced. One of the most important parameters for shielding is total cross-sectional values which were determined for 4.5

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**Table 5.** According to optimum thickness the transmission dose rates (µSv/h).

| Sample thickness (cm) | Paraffin | CC | NHC1 | NHC2 | NHC3 | NHC4 | NHC5 |
|-----------------------|----------|----|------|------|------|------|------|
| 3                     | 0.857    | 0.913 | 0.573 | 0.624 | 0.609 | 0.601 | 0.699 |
| 6                     | 0.495    | 0.561 | 0.221 | 0.262 | 0.249 | 0.244 | 0.329 |
| 9                     | 0.286    | 0.345 | 0.085 | 0.110 | 0.102 | 0.098 | 0.155 |
| 12                    | 0.165    | 0.212 | 0.033 | 0.046 | 0.042 | 0.040 | 0.073 |
| 15                    | 0.095    | 0.130 | 0.012 | 0.018 | 0.017 | 0.016 | 0.034 |
| 18                    | 0.055    | 0.080 | 0.004 | 0.008 | 0.007 | 0.006 | 0.016 |
| 21                    | 0.031    | 0.049 | 0.001 | 0.003 | 0.002 | 0.002 | 0.007 |
| 24                    | 0.018    | 0.030 | 0.000 | 0.001 | 0.000 | 0.000 | 0.003 |
| 27                    | 0.010    | 0.018 | 0.000 | 0.000 | 0.000 | 0.000 | 0.001 |
| 30                    | 0.006    | 0.011 | 0.000 | 0.000 | 0.000 | 0.000 | 0.000 |

CC: Conventional concrete NHC: New heavy concrete
MeV energy fast neutrons. The experimental equivalent dose rate of new concretes was measured and optimum thicknesses calculated for stopping fast neutrons. The results, compared with reference samples of conventional concrete and paraffin. It was concluded that new heavy concretes have an excellent radiation shielding capacity for the fast neutron and compressive strength and better high temperature resistant. Gamma radiation absorption measurements were operated for the six different gamma energies (160, 276, 302, 356 and 383 keV). Gamma-ray shielding parameters such as mass attenuation coefficients and radiation protection efficiencies were calculated. The results showed that the best radiation shielding property was found in the NHC5. It was suggested that these samples could be used in nuclear power plants to prevent radiation leaks, in storing and transporting radioactive waste, in radiotherapy rooms and also in radiation shelters.

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No potential conflict of interest was reported by the author.

ORCID

B Aygün http://orcid.org/0000-0002-9384-1540

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