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Determination of effective thermal neutron macroscopic cross-section of boron carbide samples with the help of densitometry readings using film-based neutron radiography

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Boron carbide (B_4C) is quite a unique material with respect to neutron imaging in the sense that its boron part is much better thermal neutron absorber whereas carbide offers greater scattering probability to thermal neutrons as compared to other structural materials of a nuclear reactor. Using film-based neutron radiographic technique, it is thus possible to obtain high contrast images of the subject material from where the effective thermal-neutron macroscopic cross-section ($\sum_{eff}$) can be determined with the help of densitometry readings. The transmitted part of thermal neutron flux can be estimated by the densitometry readings acquired from relatively whiter portion on an emulsion film which was occupied by the investigated sample during thermal neutron exposure whereas the incident flux is represented by the surrounding dark regions. In this paper a method is presented that can determine the value of $\sum_{eff}$ of investigated (B_4C) samples having density around $1.95 gm/cm^3$. In all the samples natural boron was used (i.e. $\approx$ 20 % ^{10}B and $\approx$ 80 % ^{11}B) along with 07 % (by weight) poly-urethane as binder. The average value of the effective thermal neutron macroscopic cross-section is found to be 0.41 cm^{-1}. In [illegible] sed on digital neutron images of the same material.

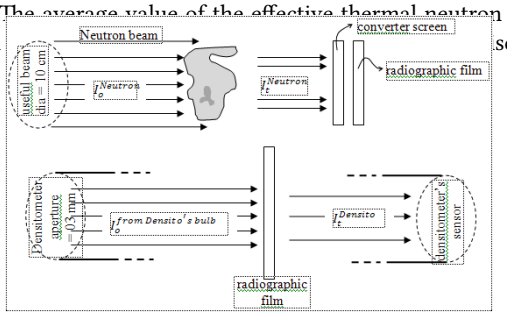

Figure 1: Image

[Please note that the radiographic film moves from the end position (in case of neutron exposure) to the centre position (in case of densitometer).]

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