

Formation of neutron fields for radiation technologies

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Abstract. The article considers the problem of forming a neutron field in the experimental channels of the water-cooled research reactors. Using the software package MCU5 we calculated a neutron field in different moderators which the experimental channel passes through. It is shown that the most appropriate moderator for neutron transmutation doping is the beryllium. The optimal location of the channel in the moderator was determined.

1. Introduction

In experiments conducted with the use of research nuclear reactors, selection of a moderator is often a decisive factor for the formation of a neutron field with the required parameters. In particular, we considered the problem of neutron transmutation doping (NTD) of silicon in water-cooled research reactors, for example, in a typical research reactor (IRT). Using the software package (MCU5TPU), we calculated a neutron field in various moderators. The experimental channel for NTD of silicon passed through these moderators. Calculations were carried out to compare the capabilities of various moderators to form the neutron field for NTD of silicon. Fig. 1 shows the "geometry" of calculations.

2. Description of reactor's construction

The core [1] consists of 20 fuel assemblies and a beryllium "trap" of thermal neutrons. The core 2 is surrounded by two reflectors 1 and 3 on four sides. The reflector 1 is made of beryllium; the reflector 3 is made of either beryllium or material of the moderator. The moderator 5 adjoins the reflector 3. The height (along the axis z) of the core, reflectors and moderator is equal to 60 (cm). In the horizontal section, the thickness of the reflector 1 is 7 (cm), the core 2 has the following dimensions: 42 (cm) (along the axis x) and 28 (cm) (along the axis y). The thickness of the reflector 3 (along the axis y) is either 7 (cm) or 14 (cm). The total thickness of the reflector 3 and the moderator 5 (along the axis y) is 63 (cm), and the width is 56 (cm) (along the axis x). All this is surrounded by water from a reactor pool.



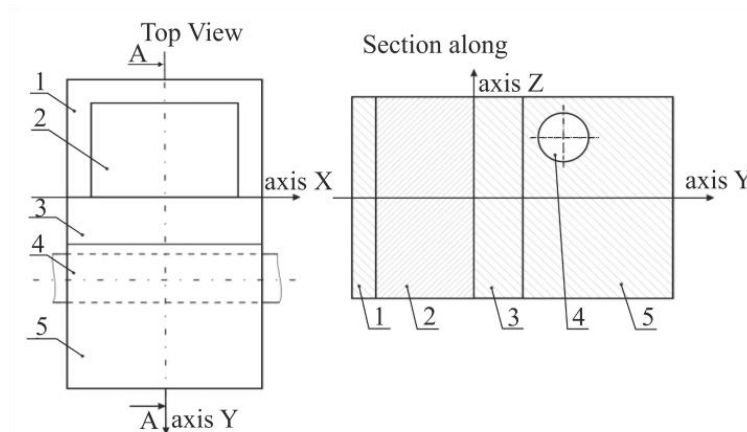


Figure 1. Scheme of a core with a reflector and a moderator:

1 – reflector from Be; 2 – core; 3 – reflector; 4 – HEC-4; 5 – moderator

In some calculations, the experimental channel is made of steel 4 with a wall thickness of 4 (mm) passed through the moderator 5. The internal diameter of the channel is 15 (cm). The channel centerline is at a distance of 12.5 (cm) from the top of the core. We considered the moderators from beryllium, heavy water and graphite. In the calculations for the core we took into account the constructional materials and coolant (water) at a distance of 12.5 (cm) from the top of the core.

Fig. 2 shows the flux distribution of thermal neutrons versus the distance to the core (along the axis y) (experimental channel is not available). We calculated the average flux density of thermal neutrons in layers of an information prism. The thickness of each layer is equal to 2.3 (cm). The total thickness of such information prism is equal to the total thickness of the reflector and moderator (63 (cm)), the height is 15 (cm), and the width is equal to the width of the moderator (56 (cm)).

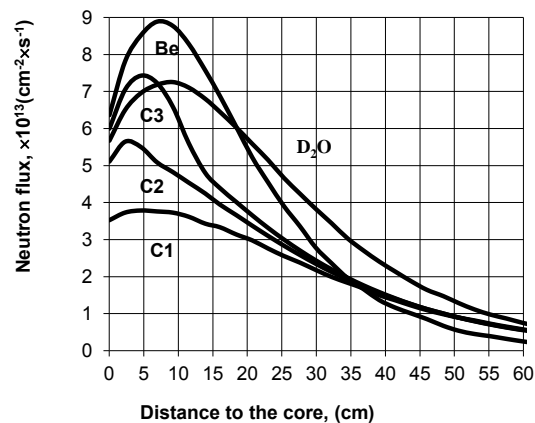


Figure 2. Flux density of thermal neutrons in different moderators

The prism axis, parallel to the side of the core, as well as a channel axis that will be considered in the further calculations, is at a distance of 12.5 (cm) from the top of the core.

The point '0' on the X-axis corresponds to the boundary of "core 2 – reflector 3 with a moderator 5" (Fig. 1). For all distributions the material of the reflector 1 (Fig. 2) was beryllium. We used the following materials as a reflector 3 and a moderator 5: distribution C1 – graphite – graphite; distribution C2 – beryllium (7 cm) – graphite; distribution C3 – beryllium (14 cm) – graphite; distribution D2O – heavy water – heavy water; distribution Be – beryllium – beryllium.

Beryllium and heavy water are more favorable moderators in comparison with other moderators in terms of the maximum flux density of thermal neutrons in the channel. However, beryllium is easy to operate and readily available for us than heavy water. Therefore, it was selected as a reflector 3 and a moderator 5 (Fig. 2).

The distribution of thermal neutron flux density along the X-axis in homogeneous reflectors of beryllium, heavy water, graphite, perpendicular to the faces of the core, has a maximum at a distance $(\tau^{1/2} - \text{age thermal neutrons})$ from the reactor core and reflectors of ordinary water – at a distance, equal to the diffusion length of thermal neutrons [3]. If this maximum will be in the reactor channel, even due to the rotation radial inhomogeneity of doping persists. Therefore, when silicon is irradiated in the tangent channel passing through the reactor's reflector, the reflector thickness between the core and the channel must be greater than the specified distance (in our opinion, 1.2-1.4 times). The other two axes distribution of neutrons in the first approximation is the cosine. The peak in the distribution of one of them (the Y-axis), perpendicular to the channel axis, should not be in the reactor channel for the same reason. The distribution of neutrons along the channel (Z-axis) affects only the axial uniformity of doping of silicon ingots. And in a translation container with bars on the channel across the irradiation zone axial doping uniformity is achieved for any distribution of the neutron channel. These considerations were taken into account in the development of the irradiation device for NTD technology at the research reactor of IRT type in Tomsk which power is 6 MW.

The results of calculations for the neutron field in the experimental channel are shown in Table 1 (L is the distance from the edge of the channel to the core; ϕ_{average} is the average neutron flux density for the length of 164 (cm); δ is the fraction of thermal neutrons in the spectrum; D is the radial flux density distribution of thermal neutrons for the diameter of 14 (cm); Q is the spectral factor (the ratio of the thermal neutron flux density to the integrated neutron flux density with an energy above 3 (MeV)). Based on these calculations, we determined the location of the channel $L = 15$ (cm) (the distance between the canal centerline and the fuel assembly is 22.5 (cm)). Table 1 shows the results of calculations.

Table 1. The results of calculations for the neutron field in the experimental channel

$L, (\text{cm})$	$\phi_{\text{average}}, 10^{12} (\text{cm}^{-2} \text{s}^{-1})$	$\delta, (\%)$	$D, (\%)$	Q
2	3.87	28.3	3.59	8.15
4	4.53	35.6	4.09	17.1
6	4.72	41.4	2.26	31.4
8	4.84	48	3.45	52.5
10	5.00	55	2.66	97.9
12	4.82	60	1.10	156
14	4.66	66	1.56	245
16	4.29	72	1.54	426
18	4.11	76	2.19	599
20	3.78	81	2.03	879
22	3.45	84	2.63	1453
24	3.08	87	2.18	1622
26	2.74	90	3.09	3442

For creating a new facility for radiation technology, it was necessary to determine the parameters of the irradiation zone. To do this, experimental researches of the characteristics of the neutron field in the HEC-4 were carried out. Activation detector method was used in measurements of the neutron field. The relative measurement technique was used to

determine the distribution of thermal neutron flux. As activation detectors the copper foil and copper wire were used, which were fixed on a special device and using a rod were fed into the reactor channel. The plotted reference points on the bar allowed to set the device in the reactor channel with an accuracy better than 3 mm, i.e. each activation detector was set at a predetermined point of the irradiation zone with an accuracy of ± 2 mm, and their mutual arrangement – with an accuracy of about 1 mm. The irradiation of the activation detector was performed at the reactor power of 20 kW for 30 minutes.

After 24-hour incubation, portions of interest were excised from the irradiated foil and wire. Then they were weighed and dissolved in nitric acid. Activity measurements were carried out in the solutions of the same geometry using the installation of γ - γ matches. In this case the relative measurement error did not exceed 3 % of relative one. The measurement results are shown in Fig. 3.

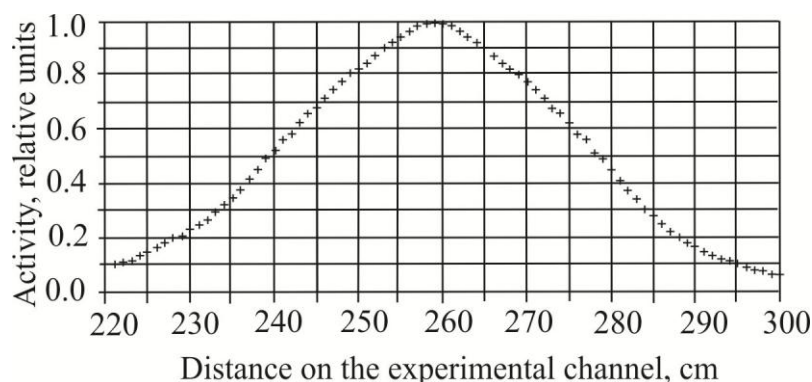


Figure 2. Distribution of the thermal neutron flux through the channel HEC-4.

3. Conclusion

Using the complex of neutron doping of silicon, we received the following parameters of the neutron field: the fraction of thermal neutrons in the spectrum is 73 %; the spectral factor (the ratio of the thermal neutron flux density to the integrated neutron flux density with an energy above 3 (MeV)) is 106; the effective temperature of thermal neutrons is 337 K; the maximum thermal neutron flux density in the channel with a irradiator is $1.5 \cdot 10^{13}$ ($\text{cm}^{-2} \text{s}^{-1}$); the average thermal neutron flux density (along the length of the reciprocating movement of the container) is $4.1 \cdot 10^{12}$ ($\text{cm}^{-2} \text{s}^{-1}$); radial irregularity of neutron fluence during rotation of the container is less than 1% . All this parameters allowed us to dope silicon with a world-class quality. Furthermore, the obtained results can be used to create other radiation technologies.

Acknowledgment

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References

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