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Contribution of neutron-capture reactions in energy release in the fuel core of BN-600

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Abstract. The use of modern computing powers and calculation methods allows to get closer to reality results of modelling, as well as to explore areas inaccessible to the experiment. Until now, the calculation of the energy released from the capture of neutrons in the reactor core has been given little attention. The method for calculation of the effective energy release components in a nuclear reactor allows to specify the values used by engineering programs for capture energy release in fast reactors. The paper presents improved method and the results of calculation of three models of the reactor BN-600. It is shown that the contribution of capture energy release in effective energy release for fresh fuel is equal to 4%, which is more than for VVER reactors. During the calculation we created a simple calculation model of the fast reactor, considering its features.

1. Introduction
Calculation of the effective energy release in a nuclear reactor is very important at the different stages of the design and operation of the reactor facility. Currently, the majority of calculation programs is not entirely correct in taking into account the capture component of the energy release, which is associated with the neutron capture reactions \((n, \gamma)\), \((n, \alpha)\), etc.). The value of the capture energy is primarily determined by the nuclide composition of the fuel core and the neutron spectrum in the reactor. Using an approximate value of the effective energy release in the reactor (associated with a drawback in the method of calculation of the capture component) may cause a significant error in the calculation of the power distribution, fuel burn-up and spent fuel characteristics. The accuracy of the calculation of effective energy is important for predicting and reconstruction of the reactors’ antineutrino spectrum [1].

The method proposed for the calculation of capture energy contribution in the total energy release in nuclear reactors based on use of high-precision calculation programs, for example, Monte Carlo codes (MCNP, MCU, SERPENT, etc.). These programs allow to conduct precise calculations and obtain results close to the experimental ones.

For the purpose of testing the method were developed three models of BN-600: two models of the equivalent cell and homogeneous model of the fuel core. To obtain nuclear reactions rates were used neutron codes MCU and MCNP. Calculations of all models were made for fresh fuel, without burn-up. All the calculations using MCNP were made in the past [2, 3].
2. Method

This method allows to calculate the mean value of the effective energy released in the fuel core of a nuclear reactor per fission. The accuracy of the obtained values depends on the used evaluated nuclear data libraries and any adopted simplifications.

Average energy release in the fission of a heavy nucleus is \( (E_f) \):

\[
E_f = E_k + E_n + E_\beta + E_{\gamma\text{ inst}} + E_{\gamma\text{ del}} + E_\nu
\]  
(1)

where \( E_k \) – kinetic energy of fission products, \( E_n \) – kinetic energy of instant and delayed fission neutrons, \( E_{\gamma\text{ inst}} \) and \( E_{\gamma\text{ del}} \) – energy of instant and delayed gamma quanta's, \( E_\beta \) – kinetic energy of delayed beta particles, \( E_\nu \) – kinetic energy of antineutrinos.

Part of fission energy which is dissipated in a nuclear reactor is called the effective fission energy \( (E_{\text{eff}}) \):

\[
E_{\text{eff}} = E_f - E_\nu - \Delta E_n
\]  
(2)

where \( \Delta E_n \) – adjustment, which takes into account loss of fission neutrons kinetic energy after absorption, which leads to fission.

Part of the energy is also released during the neutron capture reactions, considering that the effective energy release \( (E_{\text{eff}}) \) in a nuclear reactor is:

\[
E_{\text{eff}} = E_{\text{eff}} + E_{\text{capt}}
\]  
(3)

\[
E_{\text{eff}} = E_k + E_n + E_\beta + E_{\gamma\text{ del}} + E_{\gamma\text{ inst}} - \Delta E_n + E_{\text{capt inst}} + E_{\text{capt del}}
\]  
(4)

where \( E_{\text{capt}} \) – energy released in neutron-capture reactions, sum of instant capture energy \( (E_{\text{capt inst}}) \) and delayed capture energy \( (E_{\text{capt del}}) \), released during decay of products of neutron-capture reactions (is not considered in this study). The main contributors in capture energy in fuel core are \((n,\gamma)\) and \((n,\alpha)\) reactions [1,4].

The proposed method of calculation should be carried out in three stages:
1. Development of the reactor model and selection of important fission and capture reactions. Specification of the model geometry and its material composition can significantly affect the bias of the results.
2. The calculation of the rates of nuclear fission and capture reactions using developed reactor model for precise computer codes. It is necessary to take into account the fission energy dependency on neutron energy (especially for fast reactors).
3. Consistent calculation of the effective energy components, based on the rates of nuclear reactions and fission energy table of values, neutron-capture \((n,\gamma), (n,\alpha), \text{etc.}\) and decay energy yields table of values obtained from Nubase2012 and ENDF/B-VII.I [5, 6].

MCU and MCNP codes allow to obtain nuclear reaction rates in a given volume, normalized per one fission neutron. Thus, to calculate the number of fission neutrons involved in the nuclear reaction, the reaction rate should be multiplied by the average number of neutrons generated in one fission. To improve the accuracy of calculation of functionals using MCU code one should use command NBAT with a big amount of independent series (more than 10) [7].

3. Calculation of energy release in BN-600

3.1. Equivalent cell models

The first calculated model – a cell of the BN-600 reactor – is shown on figure 1 (table 1). This model is based on the description of the reactor’s fuel rod, it was developed in order to understand the
difference between the calculation of energy release components in fast and thermal reactors. Mirror reflection of neutrons is on all borders. To compensate the excessive reactivity boron absorber is put into the coolant (in the amount of 11% wt), however, the neutron spectrum is distorted and, for this reason, the model cannot be considered correct. Percentage of the integrated fission rate in the thermal (<0.625 eV), intermediate (0.625 eV - 100 keV) and fast energy regions (> 100 keV) is 0/34/66, respectively. The ratio of fissions of $^{235}$U and $^{238}$U is 87/13. The resulting value of the capture energy component is 5.9 MeV / fission, or 2.95% of the effective energy release (199.7 MeV / fission, without adjustment $\Delta E_n$) [2, 3].

In the BN-600 there are radial and end blanket zones, which strongly affect the neutron balance in the entire fuel core. Most of the neutrons disappear in radioactive capture on $^{238}$U (about 45%). For the BN-1200 the percentage of the integral fission rate in the three regions of energy equal to 11/49/40, respectively. The ratio of fissions of $^{235}$U and $^{238}$U in the BN-1000 is equal to 79/21 [8].

![Figure 1. The first and the second (with blanket) model of the BN-600 equivalent cell (cm) [9] ](image)

Method to compensate reactivity with boron was replaced by compensation with natural uranium in order to obtain a more realistic neutron spectrum (figure 1, table 1). Percentage of integrated rate of the fission reaction in the three areas of energy equal to 0/50/50, respectively. The ratio of fissions of $^{235}$U and $^{238}$U in the model is equal to 81/19. The capture energy component for the model with blanket is 7.7 MeV / fission, or 3.8% of the effective energy release (201.7 MeV / fission, without adjustment $\Delta E_n$). When calculating the effective energy release for both models was made an assumption: energy released in fission of $^{235}$U is 193.14 MeV, and of $^{238}$U is 197.7 MeV.

| Material   | Composition                      | Temperature, K | Density, g/cm³ |
|------------|----------------------------------|----------------|---------------|
| Fuel       | UO₂, 21% wt $^{235}$U            | 1200           | 8.40          |
| Cladding (steel ЧС-68) | $^{56}$Fe+2% Mo+ 15% $^{52}$Cr+10% $^{58}$Ni | 294            | 7.90          |
| Coolant    | Na (Na + $^{10}$B)*             | 500 (294)*     | 0.84          |
| Blanket**  | UO₂, 0.71% $^{235}$U            | 1200           | 8.40          |

3.2. The homogeneous model of the reactor core
The above results were tested using a closer to real-life model of BN-600 – a homogeneous model of the core with blanket zones. Its description is based on the information available in the open-source literature. The model is a cylindrical reactor core, consisting of 10 homogeneous zones (figure 2) [9, 12-16].

The actual fuel core and blanket were divided into zones, which were then homogenized: LB1 - lower axial blanket in zone with small enrichment, LB2 - lower axial blanket in zone with average enrichment, LB3 - lower axial blanket in zone with large enrichment, F1 - zone with small enrichment (17% wt $^{235}$U), F2 - zone with average enrichment (21% wt $^{235}$U), F3 - zone with large enrichment (26% wt $^{235}$U), UB1 - upper axial blanket in zone with small enrichment, UB2 - upper axial blanket in zone with average enrichment, UB3 - upper axial blanket in zone with large enrichment, RB – radial blanket. The enrichment of depleted uranium in the blanket zones is 0.3% wt $^{235}$U.

![Figure 2. Model of reactor BN-600 homogeneous core with a blanket (cm)](image)

While creating the model we did not take into account the ends of all the rods. Scram control rods and automatic control rods were considered fully withdrawn, fuel assemblies containing them were presented as a wrapping filled with coolant. The porosity of the fuel is 86% (it causes decrease in real density in comparison with the theoretical). Density of fuel was considered to be 9.426 g/cm³, density of cladding and wrapping – 7.674 g/cm³, sodium density – 0.841 g/cm³, boron carbide – 2.510 g/cm³.

Shim control rods were completely dropped in the upper blanket of small and average enrichment zones (UB1, UB2). Further, in order to make the neutron multiplication factor equal unity, they were immersed in the small and average enrichment zones (F1, F2), which resulted in the increase of atomic concentrations of boron, carbon and decrease of atomic concentration of sodium. Atomic concentrations of steel isotopes in areas with shim rods were calculated with the consideration that the steel cladding and the wrapper of shim rod put into the core on a half of its length. The critical immersion equals to 0.465 of the height of the fuel core (F1, F2).

Temperature of the lower axial blanket taken to be the same as the temperature of coolant at the reactor inlet, temperature of the upper axial blanket taken to be the same as sodium outlet temperature.
Temperature of the radial blanket equal to the average temperature of sodium in the core, temperature of the core materials was taken from BN-600 hybrid benchmark [17].

**Table 2. Temperatures of materials in BN-600 reactor model.**

| Material          | Temperature, K |
|-------------------|----------------|
| LB1, LB2, LB3     | 650            |
| F1, F2, F3        | 1500           |
| UB1, UB2, UB3     | 823            |
| RB                | 736            |

When calculating the effective energy release were defined fission reaction rates for $^{235}$U and $^{238}$U isotopes in twenty energy groups. Then were calculated fission reaction rate contributions from each energy group in the average effective energy (by weighing). Dependence of the fission energy on neutron energy was taken from ENDF / B-VII.I [6]. Similarly, the adjustment $\Delta E_n = 0.66 \pm 0.16 \text{ MeV}$ was calculated (the error is determined by the number of energy groups). It should be noted that the fission reaction on $^{235}$U caused primarily by neutrons of [0.01, 7.5] MeV energy, and on $^{238}$U – by [0.8, 8.5] MeV neutrons (contribution to the total fission rate more than 0.1%).

Since the MCU code does not allow to calculate the rates of (n, $\gamma$), (n, $\alpha$), etc. reactions separately at the full range of neutron energies, but only in the area of physical sub module FARION (fast energy region, only cross-sections for 300 K are distributed), the simplification was made: we attributed capture reaction rate to the reaction which is significantly prevail in calculation with MCNP code. Thus, we consider the capture reaction as the reaction (n, $\gamma$) for all isotopes, except $^{16}$O and $^{10}$B, which are mainly involved in the reaction (n, $\alpha$). According to our estimations, this simplification could lead to 2.5% error in the capture energy calculation. The statistical error of the Monte-Carlo method and the error in the value of nuclear reactions energy yield makes a negligible contribution to the error of capture energy component.

Percentage of integrated rates of the fission reaction in the three areas of energy equal to 0/49/51, respectively. The ratio of fissions of $^{235}$U and $^{238}$U in the model is equal to 88/12. The capture component is $8.2 \pm 0.2 \text{ MeV} / \text{fission}$ (error from simplification), or 4.1% of the effective energy ($201.4 \pm 0.3 \text{ MeV} / \text{fission}$). Comparison with previous model is in the table 3.

**Table 3. Results of effective energy release calculation for BN-600 models (MeV / fission).**

| Model                         | $\Delta E_n$ | $E_{\text{capt}}$ | $E_{\text{efe}}$ | $E_{\text{capt}} / E_{\text{efe}}$ |
|-------------------------------|--------------|-------------------|------------------|-------------------------------------|
| First (cell with boron)       | --           | 5.9               | 199.7            | 2.95%                               |
| Second (cell with blanket)    | --           | 7.7               | 201.7            | 3.8%                                |
| Third (homogeneous fuel core) | 0.66         | 8.2               | 201.4            | 4.1%                                |

As can be seen from the results, model with blanket and the homogeneous fuel core model give similar results. This suggests that the simplified model of the reactor can be used for evaluation of its neutron features and testing of energy release calculation methods. The fraction of $^{235}$U fission in full core model is higher than in the cell model with blanket. It can be explained by the fact that the blanket zone is located on the periphery of the fuel core, but not inside the core like in the cell model.
4. Conclusions
This paper presents the results of the use of previously developed method for calculation of the contribution of neutron capture reactions in the effective energy release in the fuel core of the BN-600 reactor. Three reactor models were designed and calculated (cell model, cell model with blanket and homogenous core model).

Calculation shows that the value of the capture energy release in the fresh core of BN-600 reactor (OU2 fuel) is about 8 MeV or 4% of the effective energy release in the core. One should take into account the presence of blanket zones when calculating the neutron balance in a fast reactor.

Simple model of the reactor cell with a blanket which was developed during this work can be used for estimation of effective energy release components in the BN-600 reactor. The described technique allows to obtain the exact values of the effective energy release and to estimate its dependency on fuel burn and other characteristics of the reactor.

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