Possibilities of Pin Power Density Determination

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**ABSTRACT**

The work presents a brief description of the power density determination on critical assemblies. The comparison between experimentally determined fission densities of the selected fission products is also presented. These fission products were originated during a 2.5 hour irradiation on the power level of 9.5 W in selected fuel pins of the VVER-1000 Mock-Up (LR-0 reactor). All experimentally obtained data were also calculated by Monte Carlo method. The Net Peak Area (NPA) may be used for the determination of fission density across the mock-up. This fission density is practically identical to power density.

**1 INTRODUCTION**

The pin wise neutron flux density is an important quantity, which has to be monitored during reactor operation. It can be determined by means of fission density, which can be derived using gamma spectroscopy of irradiated fuel. It assumes the proportionality between fission product activity and fission density. This assumption is met, because the disproportionality caused by various fission distributions is negligible with respect of fission yield uncertainties being about 1.5 %. More details can be found in [1].

The gamma spectroscopy method is relatively widely used. Various types of detectors can be used. In the case of scintillation spectroscopy the single peak evaluation 1596 keV from $^{140}$La is reported for the fuel long-term irradiation and cooling [2]. In the case of short-time irradiation and cooling the measurements in wide gamma energy region have been reported in [3]. The data obtained by this method have to be evaluated by comparison with the reference fuel pin and the results are evaluated as relative distributions.

The evaluation is much easier for HPGe spectroscopy due to the better resolution of HPGe detector than NaI(Tl) crystal. The fission products with half life of about some hours, like $^{92}$Sr (Figure 1), $^{91}$Sr, $^{135}$I, $^{133}$I, $^{143}$Ce, dominate in the spectra for irradiation time about several hours and a short cooling time [4]. When the irradiation on higher power lasts some days and the measurement starts after some cooling time when short living isotopes are already decayed, then lines from nuclides with half life of about some days (like $^{140}$La) dominate in the photon spectra [5]. When the irradiation lasts for months or years there are gamma lines from $^{137}$Cs and $^{95}$Zr dominating in the spectra, and also $^{95}$Nb which can be measured as well, for example [6].
2 LR-0 REACTOR

The LR-0 reactor, operated by Research Center Rez (Czech Republic), is a pool-type zero-power light-water experimental reactor. All described experiments were performed with the VVER-1000 Mock-Up. Such mock-up core consists of 32 shortened VVER-1000 type fuel assemblies with different enrichment of $^{235}$U [7]. The fuel length is optimized for the LR-0 reactor and the length of the fissile column is 125 cm. The core layout with the marked measured fuel assemblies and the fuel pins is shown in Figure 2.
3 EXPERIMENTAL AND CALCULATIONAL METHODS

3.1 Fission rate detector arrangement

The fission products were induced during a 2.5 hour irradiation on the power level of 9.5 W. The basic part of the gamma scanning system is a semiconductor HPGe coaxial detector in a streamline horizontal configuration (Ortec GEM70, resolution approximately 2.1 keV at $E_\gamma = 1333$ keV) and a multichannel analyzer DSA2000 (Canberra). The measured fuel pin is attached to the device which performs the sequential axial movement (figure 3). This device consists of a stand, a precise helix, a stepping motor, a coupling and a gripping of the fuel pin. The collimator tube consists of a central part and a surrounding Pb shielding. The central part is a lead cylindrical pipe with a length of 10.14 cm, an outer diameter of 8 cm. In this pipe there is a collimator window centrally located with respect to the HPGe crystal. The window width is 1 cm in collimator back end cap to cover the pin diameter being 0.9 cm. The window in HPGe front end cap is 4.9 cm to maximize the detector efficiency. The several variants of this central part vary in the height which enables wide range of applications. The higher collimator (4.9 x 1 cm) is suitable for a short time measurement while the smallest one (2 x 1 cm) is suitable for a long time measurement. The small collimator can be used for both radial and axial measurement while the higher one is intended for the radial profile measurement. During the radial profile measurement, the measurement was performed in the centre of the pin fissile column, lying in the central horizontal plane of the reactor core. That means, in the case of 4.9 cm height collimator, 60.05 cm above the bottom of the fissile column to 60.05 cm below the top of it. Figure 3 shows schematically the cross-sectional diagram of the collimator tube used in the gamma-ray spectroscopy measurements. The measured gamma spectra were analyzed with the Genie 2K software (Canberra).

![Figure 3: The layout of the experimental arrangement](image)

3.2 Activation foil detector arrangement

The absolute value of the Mock-Up power was determined by means of the reaction rate in the activation foil using the reaction $^{197}$Au(n,g)$^{198}$Au. The gold in mass about 30 µg per foil was in an Aluminium alloy with 1 % of Au. The foil diameter was 3.6 mm and its thickness was 0.1 mm. The $^{198}$Au peak at 411.8 keV was measured immediately after the irradiation by the coaxial HPGe detector in the vertical configuration (Ortec GEM35). The experimental
uncertainties were estimated to be 1 - 2.5 % due to the low amount of gold and therefore the relatively low induced activities.

Simultaneously, a monitoring of the reactor power was carried out by an independent measurement with a compensated boron-lined chamber placed in the lateral reflector region of the reactor core. The discrepancies between the measured reaction rates of gold, normalized to the response of the chamber, were lower than 2.5 %. It can be concluded on the basis of such agreement of these independent methods that the uncertainty in the location of the activation foil during the irradiation does not play any role.

3.3 Calculation methods

The pin-by-pin power density distribution in the reactor core was calculated by the MCNPX transport code [8] with the use of ENDF/B VII nuclear data library [9]. The calculation covers all pins in the VVER-1000 Mock-Up core but the comparison is realized only for the selected pins. The benchmark value of 4.6 ± 0.1 g/l of H$_2$BO$_3$ diluted in the moderator was used during the calculation [7]. A detailed sensitivity analysis shows that the effect of the uncertainty in the boric acid on the fission density uncertainty is lower than 0.8 %. This value is comparable with the MCNPX statistical uncertainty and it is notably lower than pulse rate uncertainties.

The photon transport, namely HPGe responses, was also determined using the MCNPX code. The parameters of the detector provided by the producer were used for calculation. An experimental absolutization of the apparatus, which would allow an absolute determination of the activities of the selected fission products, is very difficult, if not impossible, due to the complex geometry of the source and mainly due to the absence of the standard with relevant gamma lines. For this reason, the comparison was done directly for the response of the HPGe normalized to the irradiation power during the first step (9.5 W).

It should be noted that the values of the fission yields are affected by appreciable uncertainties, resulting in an uncertainty of production of fission products related to the fission per unit with value approximately 1.5 %. The mechanism of their creation is both direct and sequential by beta decays (and beta decay and delay neutron emission) of their mother nuclides. The scheme of this creation is described below.

3.4 Fission products

Due to the suitable values of emitted ray energies, their half-lifes and the activity after defined irradiation conditions and also due to little differences between the expected and measured decay, $^{92}$Sr fission product was chosen for the determination of the absolute power density in selected pins. More details about this experimental method can be found in [10].

The energy spectrum of neutrons in various regions of the mock-up is not homogenous. It is harder in the region adjacent to the baffle and softer near gap regions. This influence is not as significant as it may seem, however, because the fluctuations in the production of $^{92}$Sr caused by the fission induced by fast, resonant or thermal neutrons are lower than 0.5 %.

\[ ^{92}\text{Se} \rightarrow ^{92}\text{Br} \rightarrow ^{92}\text{Kr} \rightarrow ^{92}\text{Rb} \rightarrow ^{92}\text{Sr} \]

The paper presents also results from evaluation of $^{97}$Zr decay peak located at 743.4 keV. The energy of the gamma photons from $^{134}$Te decay is about 1 keV smaller than from the $^{97}$Zr one. The system is not able to distinguish between these two peaks due to HPGe detector’s 1.7 keV resolution half-width. The observed peak is formed by both $^{97}$Zr photons with energy 743.36 keV and $^{134}$Te photons with energy 742.4 keV.
For the evaluation of experimentally determined NPA (net peak area), the results have to be corrected because of the radioactive decay of the gamma emitters in the irradiated fuel pins. The correction is relatively easy to perform, if the peak is formed only by one nuclide, and the mother nuclei has got a notably smaller half life. If not, and the measured peak has got more sources of the same photon energy, there are two ways of decay correction. The measured peak is purged from parasitic peaks in this paper. The presence of natural Zr in fuel elements might distort the experimental data as well. The $^{97}$Zr may origin not only during fission but also by activation of the cladding tube. The Mock-Up fuel element consists of UO$_2$ pellets in a zirconium-alloy cladding. Both the calculation estimates and the experimental ones show that the activation of the fuel cladding tube ($^{96}$Zr) adds about 3.39% to the total $^{97}$Zr value measured by means of 743.36 keV photons. More details can be found in [10].

$$\text{97} \text{Kr} \rightarrow \text{97} \text{Rb} \rightarrow \text{97} \text{Sr} \rightarrow \text{97} \text{Y} \rightarrow \text{97} \text{Zr}$$ (2)$$

$$N_i^{Zr-97}(t) = N_i^{743.36 keV}(t)K_i(t) ; N_i^{Zr-97}(\text{fission}) = N_i^{Zr-97}(1-C).$$ (3)

The $^{135}$I fission product is interesting due to its relatively long half live and relatively high energy of the emitted photon. It has to be noted, that this is the right isotope which is responsible for the xenon poisoning, because $^{135}$I decays to $^{135}$Xe with the 2.66x10$^6$ b absorption cross section.

$$\text{135} \text{Sn} \rightarrow \text{135} \text{Sb} \rightarrow \text{135} \text{Te} \rightarrow \text{135} \text{I}$$ (4)

<table>
<thead>
<tr>
<th>Table 1: Main evolution chains of various fission products [8]</th>
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<tbody>
<tr>
<td>$^{92}$Se</td>
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<td>$^{92}$Br</td>
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<td>$^{92}$Kr</td>
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<tr>
<td>$^{92}$Rb</td>
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<tr>
<td>$^{92}$Sr</td>
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4 RESULTS AND DISCUSSION

All results are presented in the form of C/E-1 comparison in Figure 4. The calculation was performed using the ENDF/B-VII. Measured pins are highlighted in the core layout in Figure 2. The pins were selected especially in the regions where the $^{97}$Zr NPA is sufficiently great, i.e. pins in the central parts of Mock-Up. The uncertainty plotted in Figure 4 by the grey line covers the most notable sources of the uncertainty. It is specifically the MCNPX statistical uncertainty (bellow 1.7%), the uncertainty in fission product yields (about 1.5%), the uncertainty of the fission product measurements (bellow 1.7%), the uncertainty in the position of the activation foil (about 1%), the uncertainty in the measurement of the activation foil (bellow 2.5%) and the uncertainty in the boric acid density (bellow 0.9%), as the most notable material parameter. It can be assumed that all the uncertainties are non-
correlated, thus the total uncertainty is square root of the sum of the squares of all combined uncertainties.

![Figure 4: C/E comparison for selected pins NPA of selected fission products](image)

**4.1 Determined fission density across the benchmark**

The comparison between calculated and experimentally determined NPA is precise comparison. The disadvantage is that it does not allow to easily compare the fission density determined by various fission products. The detailed comparison of the fission density determined using various fission products is presented in Figure 5. It is worth noting the results of all presented nuclides are in satisfactory agreement with the experiment.

![Figure 5: Determined fission density in the Mock-Up](image)
5 CONCLUSIONS

The results are in satisfactory agreement with the calculations. The best agreement can be found in the case of $^{92}$Sr. $^{97}$Zr is also suitable nuclide for the power determination of zero power reactors. Its main advantage is its half life, which enable measurement of a relatively large amount of fuel pins in one irradiation batch. In the case of measurement carried out between 11 to 38 hours after the reactor shutdown, the uncertainty increases not more than 1 %. The disadvantage with regard to $^{92}$Sr is lower NPA, originating from the lower decay constant, thus requesting a longer measurement time than $^{92}$Sr.

The Iodine has got higher energy of emitted photons, thus its detection is easier than of $^{97}$Zr because there are less peaks. On the other hand its half life, even much longer than $^{92}$Sr, is notably smaller than $^{97}$Zr.

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