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#04-80 Sensitivity Analysis of an Advanced Measurement Method for Thermal Neutrons Absorbers Detection in Irradiated Beryllium

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Operation experience in many Material Testing Reactors, where beryllium moderator or reflector is used, has shown that accumulation of neutron absorbers, so-called poisons, has a non-negligible impact on both core performance and safety aspects due to n-Be interactions, and needs to be addressed properly in calculations. As a result, depletion of beryllium, called also beryllium poisoning, was implemented in core follow-up deterministic calculations in several facilities. However, Material Testing Reactors are designed mainly for performing irradiation tests of material behaviour in severe conditions, resulting in frequent repositioning of the core elements in order to host experiments. This specificity makes almost impossible to follow the exact irradiation history of each beryllium elements and consequently inferring the exact accumulation of poisons. Thus, there is a strong need for developing a methodology enabling measurement and assessment of beryllium poisoning.

We addressed the problem of quantifying poisons accumulation using neutron transmission method. It is based on measuring transmission and absorption of thermal neutrons in beryllium elements. The measurement set is composed of a neutron source, a neutron moderator and a neutron detector, mounted on a bench, with beryllium sample placed between detector and moderator. In the study, we used the MARIA reactor's beryllium moderator blocks as material samples. Their characteristic truncate shape contributes to uneven axial neutron flux distribution in the core. To enable evaluation of axial distribution of poisons, both neutron source and detector move simultaneously along the block. A non-irradiated beryllium block has been measured and constitutes a reference point. We then performed series of irradiations in the NCBJ's MARIA reactor on several beryllium blocks delivered in the same batch. Comparing measurement results of irradiated blocks to the reference results, allows evaluating thermal neutrons absorbed in irradiated beryllium. Conducting such experiments requires taking into account several effects that might affect the measurement accuracy. In order to evaluate these effects, we performed a sensitivity analysis and all the above steps were simulated with the SERPENT2 Monte-Carlo code, using JEFF3.1.1 nuclear data library. We focus on analysis of expected uncertainties related chosen detector, impurities in beryllium, nuclear data uncertainty and gamma activation of irradiated beryllium.

In our study, we analysed the possibility of using following detectors: fission chamber with 235U deposit, 3He detector, 10B lined-detector. As no pure thermal neutron source exists, polyethylene-moderated neutron sources have been tested to be used for measurements, taking into account characteristics of the system. Neutron sources considered in this study include Am-Be, Pu-Be, Sb-Be, 252Cf. For each of these sources, we studied the optimal polyethylene thickness.

In this paper, we present the newly developed method for measurement of these neutron absorbers'concentration, we discuss calibration and sensitivity study of the above described measurement setup. The methodology and codes used for simulations are described. The results of the optimization study are detailed, and choice of specific type of detector and neutron sources are justified. Summarizing the results, we also give some recommendations.

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