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Application of dosimetry measurements to analyze the neutron activation of a stainless steel sample in a training nuclear reactor

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HIGHLIGHTS

- Neutron activation of materials in the core of a nuclear reactor.
- Application of the Monte Carlo method to simulate neutron activation.
- Importance of steel components of the reactor core for neutron activation.
- Irradiation of a stainless steel sample in a nuclear reactor.
- Doses measured around the sample are compared with MC simulation results.

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ABSTRACT

All materials present in the core of a nuclear reactor are activated by neutron irradiation. The activity so generated produces a dose around the material. This dose is a potential risk for workers in the surrounding area when materials are withdrawn from the reactor. Therefore, it is necessary to assess the activity generated and the dose produced. In previous works, neutron activation of control rods and doses around the storage pool where they are placed have been calculated for a Boiling Water Reactor using the MCNP5 code based on the Monte Carlo method. Most of the activation is produced indeed in stainless steel components of the nuclear reactor core not only control rods. In this work, a stainless steel sample is irradiated in the Training Reactor AKR-2 of the Technical University Dresden. Dose measurements around the sample have been performed for different times after the irradiation. Experimental dosimetric values are compared with results of Monte Carlo simulation of the irradiation. Comparison shows a good agreement. Hence, the activation Monte Carlo model can be considered as validated.

1. Introduction

All materials present in the core of a nuclear reactor can be activated by neutron irradiation. When activated materials are withdrawn from the reactor, a dose is produced around them. This dose is a potential risk for workers and people staying in the surrounding area. Therefore, it is necessary to assess the activity generated and the dose produced.

In previous works (Ródenas et al., 2010a, 2010b, 2010c, 2010d, 2010e), neutron activation of control rods and doses around the storage pool where control rods are placed have been calculated for a Boiling Water Reactor using the MCNP5 code (Monte Carlo Team, 2003) based on the Monte Carlo method.

On the other hand, most of the activation is produced in stainless steel components of the control rod. Indeed, many components in the nuclear reactor core are made of stainless steel. Therefore, the Monte Carlo model can be applied to the activation produced in a piece of stainless steel exposed to some neutron flux in a reactor. The dose rate around the activated piece can be measured as well.

In this work, a stainless steel sample is irradiated in the Training Reactor AKR-2 (Hansen and Wolf, 2009) of the Technical University Dresden. Dose measurements around the sample have been performed for different times after the irradiation.

Experimental dosimetric values are compared with results of the Monte Carlo simulation and the comparison shows a good agreement. Activities obtained with the Monte Carlo model of the neutron activation are used as input data for the second Monte Carlo model simulating the dose produced around the irradiated

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piece. These doses are compared with dosimetry measurements. As comparison shows a good agreement between measured and simulated doses, the activation Monte Carlo model can be considered as validated.

2. Methodology

2.1. Neutron activation

The activity generated in neutron reactions depends on reaction cross sections, neutron spectrum, neutron flux distribution, concentration of precursors of each radionuclide, and irradiation time. After irradiation, activities decrease with time and disintegration constants.

The interaction rate Q (reactions/cm^3 s) is given by

$$Q = C \int \Phi(E)\sigma(E)\mathrm{d}E \tag{1}$$

where C is the atom density (at/b cm) depending on the target concentration; Φ(E) is the neutron flux (n/cm^2 s); and σ(E) is the microscopic cross section of the reaction (b). On the other hand, for each j-isotope generated, a matter balance can be done as follows:

$$\frac{\mathrm{d}N_j}{\mathrm{d}t} = Q_j - \lambda_j N_j \tag{2}$$

integrating, the concentration N_j (nuclei/cm^3) of the j-isotope is obtained, with t_i being the irradiation time:

$$N_j(t) = \left(\frac{Q_j}{\lambda_j}\right)(1 - e^{-\lambda_j t_i}) \tag{3}$$

For a cooling time t_c the concentration N_j becomes

$$N_j(t) = \left(\frac{Q_j}{\lambda_j}\right)(1 - e^{-\lambda_j t_i})e^{-\lambda_j t_c} \tag{4}$$

and multiplying by λ_j to obtain activity:

$$A_j(t) = Q_j(1 - e^{-\lambda_j t_i})e^{\lambda_j t_c} \tag{5}$$

$A_j(t)$ is a volumetric activity (Bq/cm^3). To obtain the total activity it is necessary to multiply by the sample volume. The maximum activity will be the saturation activity, Q_j, asymptotic value considering an irradiation time very long and neglecting the cooling time.

2.2. Experimental measurements

The Training Reactor AKR2 (Hansen and Wolf, 2009), acronym for Ausbildungskernreaktor 2, is located at the Technical University in Dresden, Germany. It is a zero power, thermal reactor moderated by solid polyethylene. The fuel elements consist of a homogeneous mixture of moderator and uranium oxide fuel enriched 19.8%. It has a maximum power of 2 W and the maximum neutron flux in the central experimental channel is Φ_{max}=5E+07 n/cm^2 s.

The active zone of the core is made up of disk shaped fuel elements with a diameter of 25 cm. The height of the active zone is 27.5 cm.

For the experiment, the reactor is driven at a power level of P=0.59 W. This corresponds to a measured neutron flux of 2.5E+07 n/cm^2 s. This flux has been measured in the central experimental channel of the reactor. The cross section of the whole reactor is shown in Fig. 1.

A stainless steel sample type X8CrNiTi18.10 is irradiated for 10 h in the central experimental channel of the reactor. The sample has a cylindrical shape with a radius of 1 cm and a length of 7 cm. It has a volume of 21.99 cm^3 and a density ρ=7.9 g/cm^3. The composition of the sample is listed in Table 1 (Thyssen Krupp Materials International, 2007).

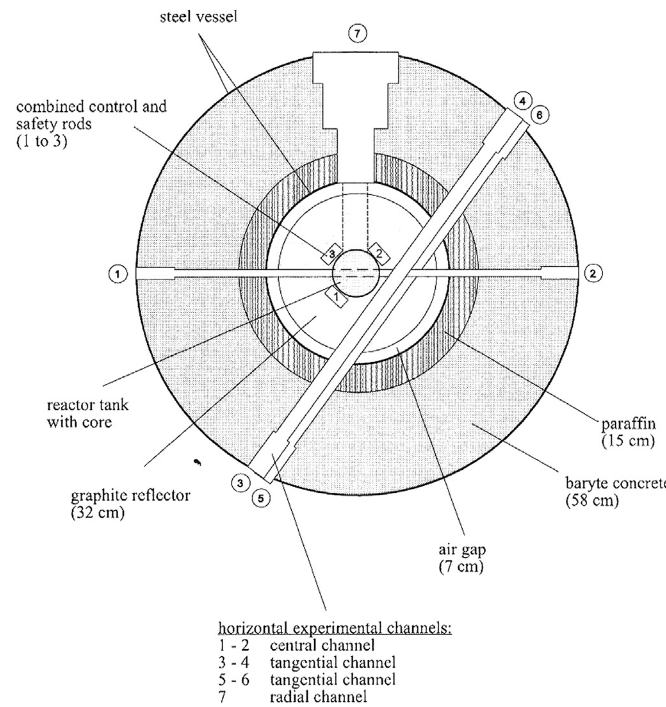

Fig. 1. Cross section of the whole reactor.

Table 1
Composition of the sample.

Element	Weight fraction (%)
Cr	19.000
Ni	12.000
C	0.100
Si	1.000
Mn	2.000
P	0.045
S	0.015
Ti	0.400
Fe	65.440

Dose experimental measurements were performed with a Berthold dose rate meter type LB133-1 equipped with an ionization chamber detector LB6006, suitable for photon dose equivalent measurements in photon energy range 30 keV–1.3 MeV. The device was calibrated by official authorities.

2.3. Monte Carlo model

An activation Monte Carlo model has been developed using MCNP5. The interaction rate Q (Eq. (1)) is calculated using F4 tally and FM4 (tally multiplier card), which provides data for the reactions included in the calculation, listed in Table 2.

The energy spectrum of fission neutrons (Lamarsh and Baratta, 2001) used for the simulation is the Watt distribution described by Eq. (6).

$$\chi(E) = 0.453e^{-1.036E} \sinh(2.29E)^{1/2} \tag{6}$$

The Watt fission spectrum can be considered as a Maxwellian spectrum from a moving reference system (Froehner and Spencer, 1980). The Maxwell fission spectrum alone describes the energy