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# DOSIMETRY OF SURVEILLANCE SPECIMENS IRRADIATED AT WWER-1000 REACTOR

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Main principles of operation of program package MCSS are presented. This package is a main part of methodology for determination of neutron flux functionals on surveillance specimens irradiated at WWER-1000 reactor, developed by specialists of INR of NAS of Ukraine. Improved surveillance program realizing at present time at the WWER-1000 reactor of Rovno NPP Unit 4 are described briefly.

## Introduction

To control reactor pressure vessel materials state the surveillance program is carried out at operating Ukrainian NPP's with WWER-1000 reactors. Round container assemblies (CA) with surveillance specimens (SS) are set on majority of units above reactor's baffle, as it's shown at Fig 1. Therefore SS are irradiated in neutron field with large neutron flux gradients.

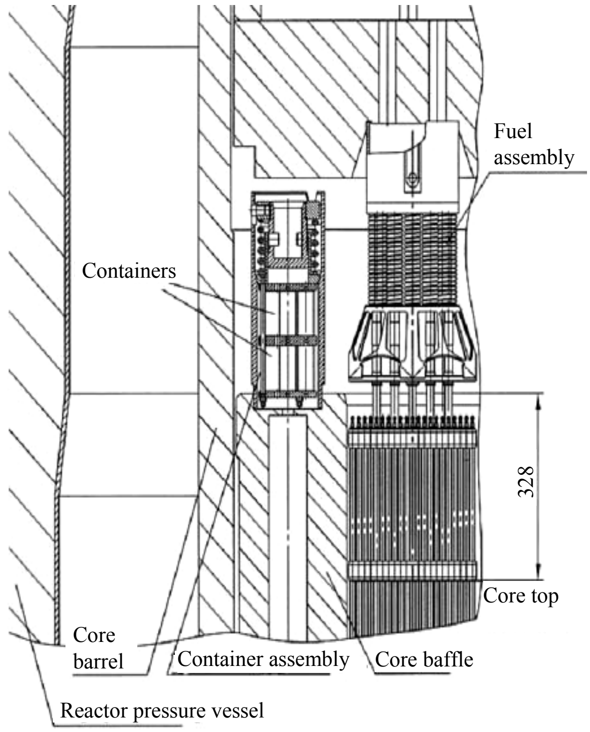


Fig. 1. Location of container assemblies in WWER-1000 reactor.

At Fig 2 the container with Charpy specimens is presented. As it is shown capsule with neutron flux indicators and V-notch are located at different places. Furthermore, a lot of containers have no capsule with neutron flux indicators. Thus, dosimetry part of SS program doesn't allow obtaining reliable information on surveillance specimens irradiation conditions. Therefore in our institute we created and update permanently the methodology of obtaining of this important information.

## The methodology for determination of neutron flux functionals on surveillance specimens

The most important items of our approach are:

calculation in the detailed 3D reactor model by Monte-Carlo method;

taking into account the changes of core condition for every 40 effective days;

full core calculation for preparation of neutron source parameters in 9 (of 30) core layers in assemblies of four periphery rows;

approximation of axial power distribution for 300 core layers;

taking into account the changes of neutron source parameters because of burn-up by WIMS-D4 calculations of used fuel assemblies;

using of calculational-experimental technique for determination of container assembly orientation with respect to the core;

taking into account the individual parameters of given reactor.

## Neutron flux calculation in WWER-1000 container assemblies

The methodology is based on program package MCSS [1] designed by us. The transport program of package is used the Monte-Carlo method for modeling the physical process of neutron transport in the calculational reactor model (CRM). CRM is based on WWER-1000 design documentation and imitates in detail the reactor construction elements that influenced on calculation results of neutron transport to SS locations. In this model the fact of the absence of water gap between core and baffle in the reactor of South-Ukraine NPP Unit 1 is taken into account. The cross-sectional and vertical views of CRM are shown at Fig. 3.

CRM includes detailed CA model (Fig. 4). Calculational detectors are located at the level of centers of Charpy specimens and centers of their upper and lower halves.

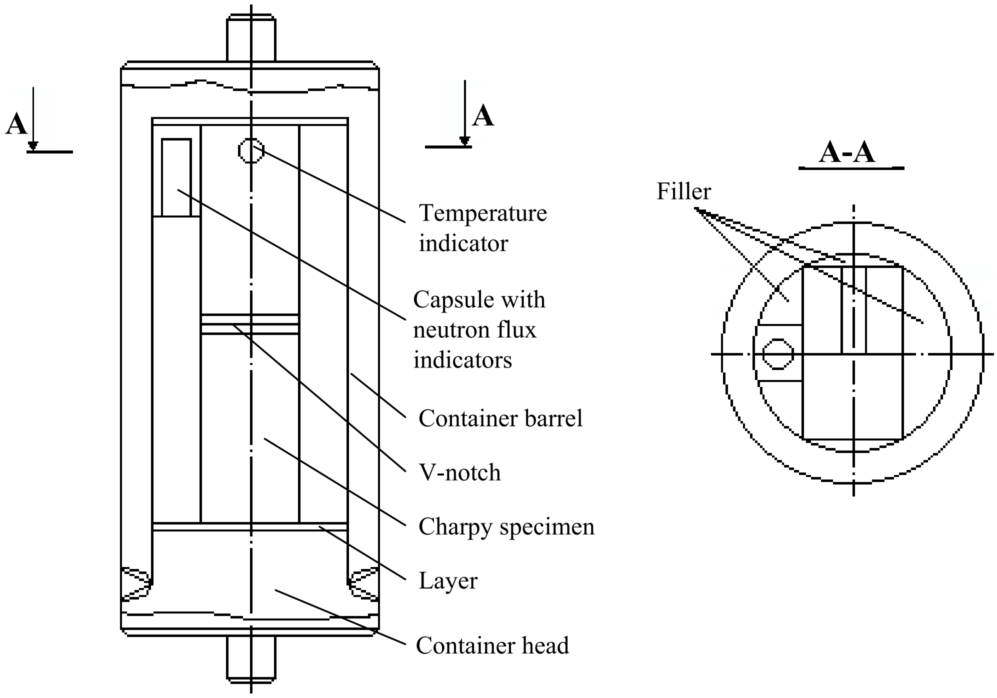


Fig. 2. Container with Charpy specimens.

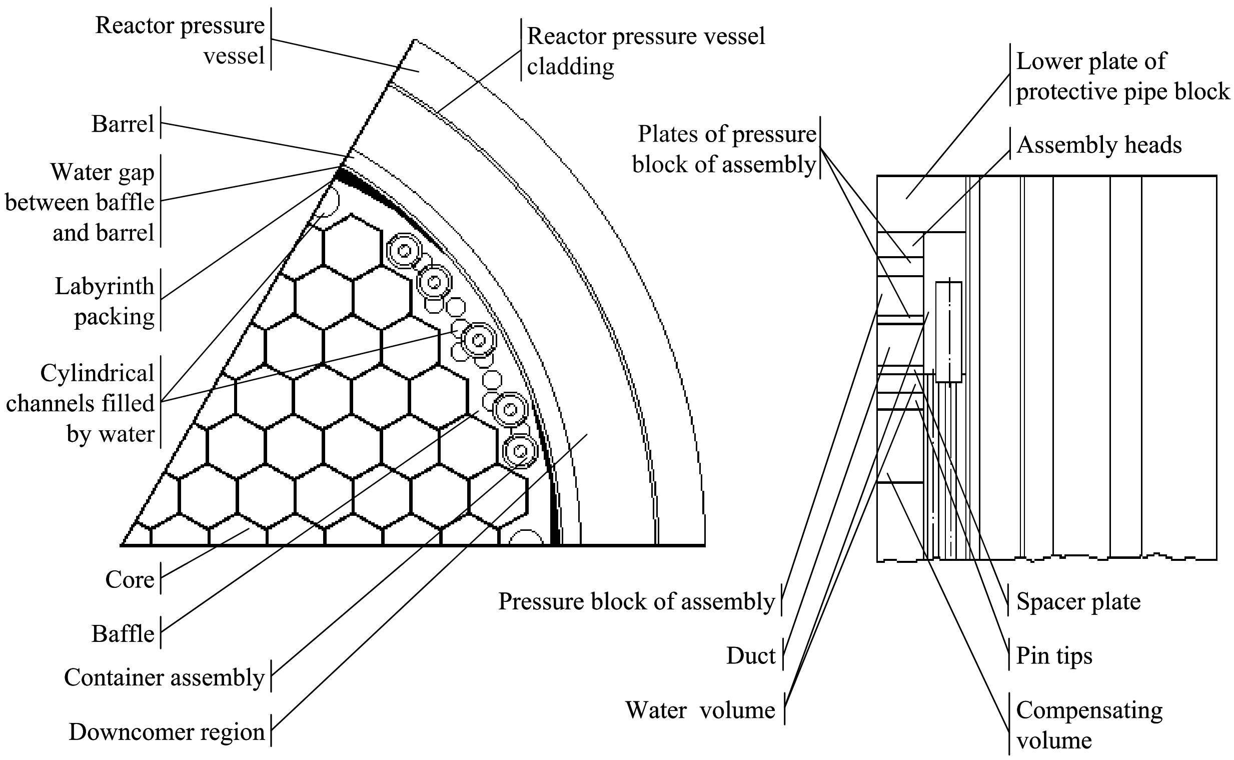


Fig. 3. Calculational reactor model.

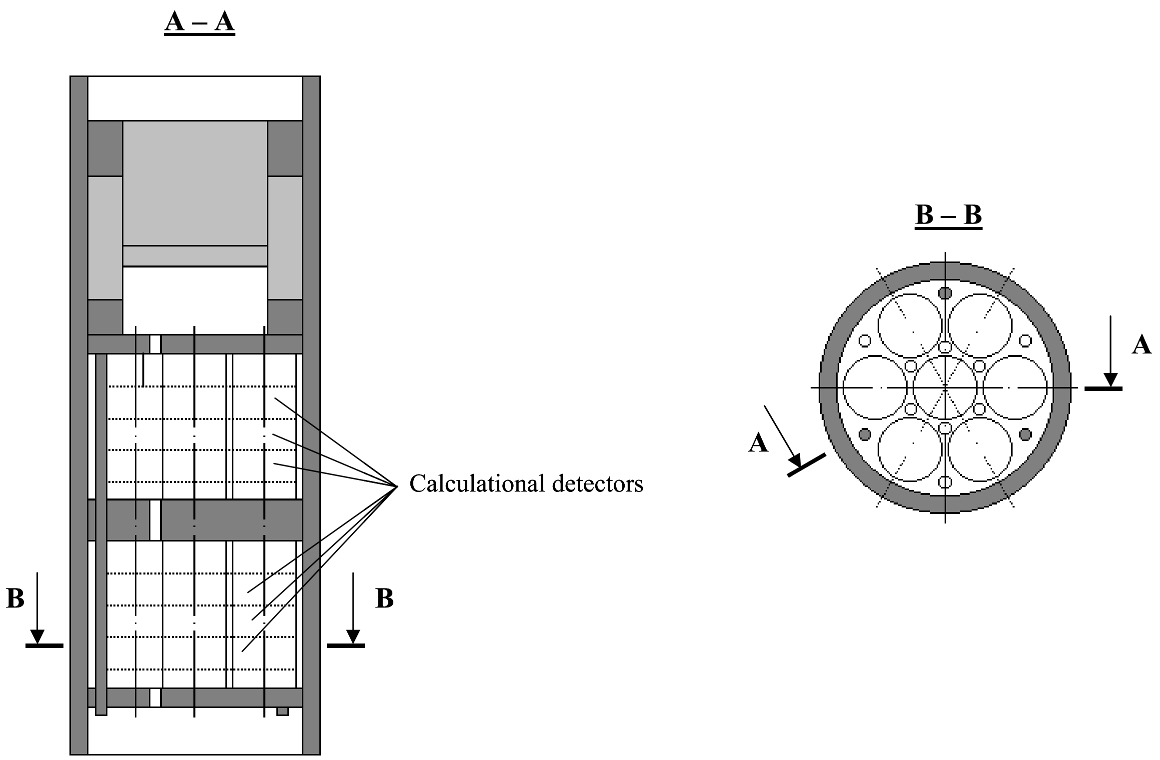


Fig. 4. Container assembly model.

**Neutron source parameters**

We prepare the neutron source parameters for transport calculation basing on full core calculation for every 40 effective days by БИПР and ПЕРМАК programs.

The first neutron source parameter is neutron yield

$$S = q\sum_j \nu_{fj}(B), \tag{1}$$

where amount of energy released in the present source

$$q = k_v k_k \frac{Q}{K^S}, \tag{2}$$

where $k_v$, $k_k$ – relative power coefficients calculated by БИПР and ПЕРМАК programs correspondingly; $Q$ – reactor thermal power; $K^S$ – source number.

The second neutron source parameter is mean group contributions to normalized source spectra

$$\chi_g = \frac{\sum_j \nu_{fj}(B)\chi_{gj}}{\sum_j \nu_{fj}(B)}, \tag{3}$$

where $\chi_{gj}$ – mean group contributions to normalized source spectra of $j$-th isotope.

In formulas (1) and (3) $\nu_{fj}(B)$ is mean amount of neutrons forming at energy unit released due to fission of $j$-th isotope nuclei in $f$ type fuel assembly depending on burn-up $B$.

Values of function $\nu_{fj}(B)$ for isotopes $U^{235}$, $U^{238}$, $Pu^{239}$ and $Pu^{241}$ in the range of fuel burn-ups from 0 to 50 MW days/kg with step of 1 MW days/kg were obtained by code WIMS-D4.

### Approximation of axial power distribution

The program БИПР allows dividing the core only on 30 layers. It's not enough. Therefore we develop the ideology of approximation of axial power distribution.

The main ideas of this ideology are next.

Mean power values $p_i$ for $n$ given layers along core height

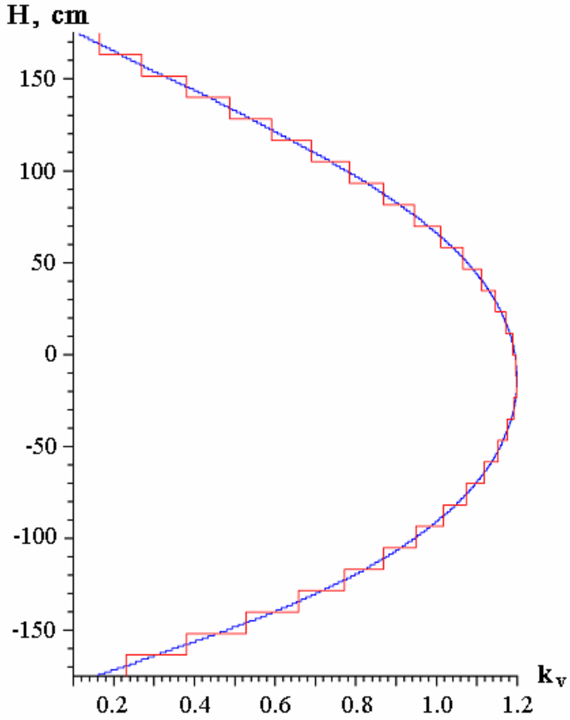


Fig. 5. Axial power distribution.

$$p_i = \frac{1}{\Delta z}\int_{z_i}^{z_{i+1}} p(z)dz\ ,\ i = 1,\ldots, n. \quad (4)$$

Boundary conditions should be given in a following way.

Logarithmic derivative at core lower end-wall

$$\frac{1}{p}\frac{dp}{dz}(z_1) = \gamma_1 . \quad (5)$$

Logarithmic derivative at core upper end-wall

$$\frac{1}{p}\frac{dp}{dz}(z_{n+1}) = \gamma_{n+1} . \quad (6)$$

The polynomial

$$p(z) = a_i + b_i z + c_i z^2,\quad z \in [z_i, z_{i+1}],\quad i = 1,\ldots,n\ , \quad (7)$$

is used beginning with the statement of power distribution function smoothness along core height. Coefficients $a_i$, $b_i$ and $c_i$ ($i = 1,\ldots, n$) are determined starting from given values $p_i$, continuity conditions $p(z)$ and its first derivative at boundaries of intervals $[z_i, z_{i+1}]$, as well as given (in БИПР input data) values $\gamma_1$ and $\gamma_{n+1}$.

At Fig. 5 our approximation method of axial power distribution is shown. The red line on the graph corresponds to distribution obtained for 30 core layers. The blue one reflects the axial power distribution for 300 layers determined using the approximation approach described above. It is important to mention that using of such an approach goes to decrease of calculated flux on SS to 5 - 7 %.

### Technique of determination of container assembly orientation

There is no reliable information about CA orientation with respect to the core. Therefore the calculational-experimental technique of CA orientation determination was developed.

Taking into account the CA location in reactor and core configuration distribution of linear neutron flux functionals can be described with this formula

$$A(\alpha) = \bar{A} + A'\cdot\cos(\alpha - \alpha_{\mathrm{m}}) = \bar{A} + A_{\mathrm{c}}\cdot\cos\alpha + A_{\mathrm{s}}\cdot\sin\alpha\ . \quad (8)$$

Values of parameters $\bar{A}$, $A_{\mathrm{c}}$ and $A_{\mathrm{s}}$ are obtained by minimization of functional

$$F = \sum_{i=1}^{N}\left(\frac{A_i - A(\alpha_i)}{\Delta A_i}\right)^2 , \quad (9)$$

where $A_i$, $\Delta A_i$ – value of considered functional and random component value of its uncertainty at point $\alpha_i$ .

Basing on these three parameter values we obtain the angle $\alpha_{\mathrm{m}}$, which is necessary for the determination of container assembly orientation with respect to the point of maximal value of considered neutron flux functional as it's shown at Fig. 6.

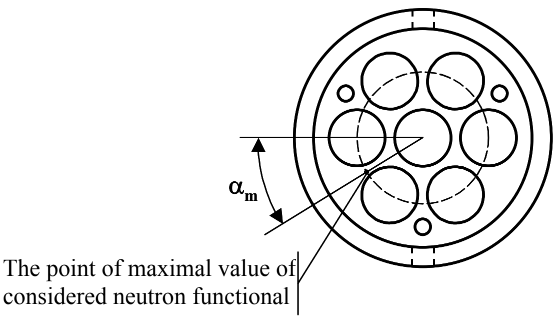


Fig. 6.

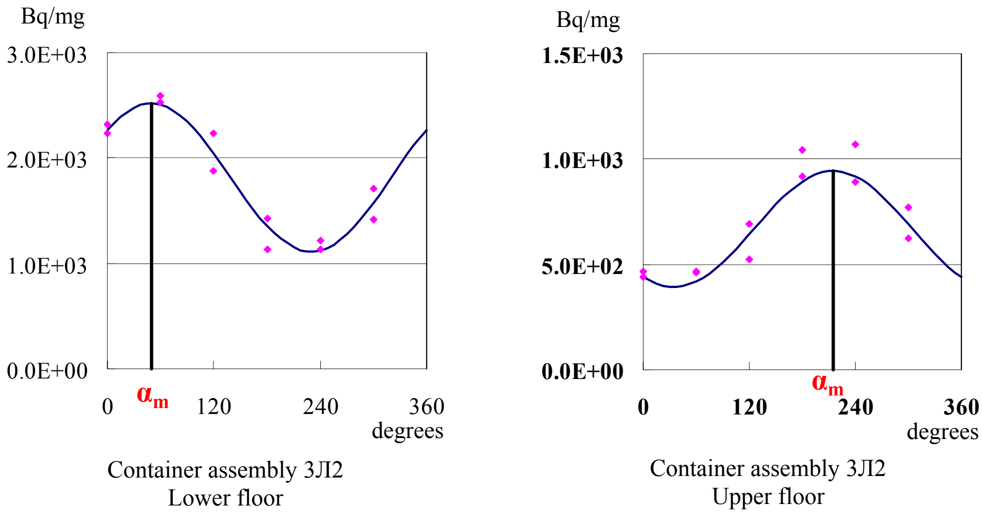


Fig. 7. Dependence on angle α the activities of $Mn^{54}$ in cuts from V-notch area of specimens irradiated at South-Ukraine NPP Unit 1 reactor.

Using calculation we obtain the angle between directions to the point of functional maximal value and to the core central axis. Thus we determine CA orientation with respect to the core.

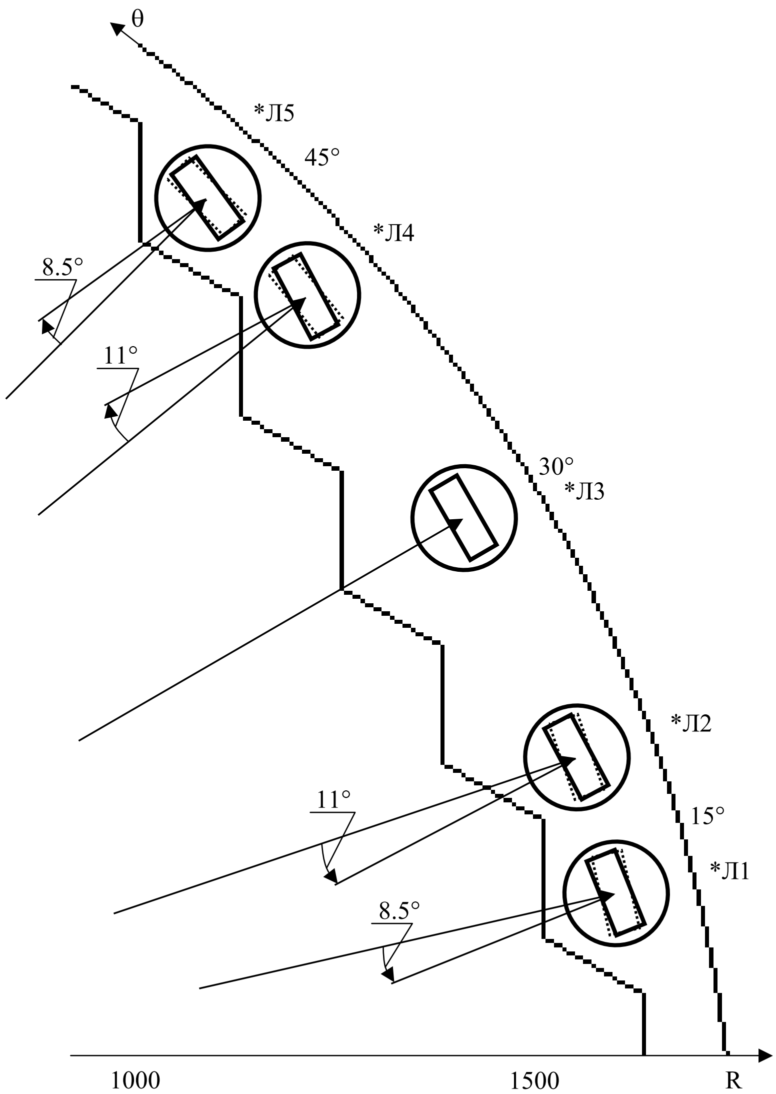


Fig. 8. Scheme of container assembly orientation.

The graphs at Fig. 7 demonstrate a close coincidence between distribution described by formula (8) and experimental data obtained for the activities of $Mn^{54}$ in cuts of SS irradiated at South-Ukraine NPP Unit 1 reactor.

The points correspond to experimental $Mn^{54}$ activities in cuts. The black curves reflect the activity distribution described by formula (8). Basing on these curves we obtain the angles $\alpha_m$ necessary for CA orientation determination.

### Improved surveillance program realizing at the WWER-1000 reactor of Rovno NPP Unit 4

To eliminate known shortcomings of regular surveillance program it was decided to improve this program for putting into operation Rovno NPP Unit 4 [2]. As a result of modernization some shortcomings were eliminated because:

SS were located in plane containers;

CA orientation with respect to the reactor core was optimized (Fig. 8) to decrease flux gradient on working parts of SS irradiated in the same container;

dosimetry part of surveillance program was improved.

There were installed the lengthy neutron activation detectors (NAD) in the form of wires from metallic iron, niobium, copper in all containers of both upper and lower floors of each CA for determination of neutron fluences on SS working parts.

Lengthy NAD were installed in aluminum fillers of containers on the levels of SS working parts and also in the space formed by V-notches of adjacent Charpy SS. In addition NAD were located in the middle of both upper and lower halves of Charpy and COD SS. It allowed the determination neutron fluences on working parts of reconstructed SS in case of SS reconstruction technology applying.

REFERENCES


1. *Bukanov V.N., Dyemokhin V.L., Gavriljuk V.I. et al.* Overview of the Surveillance Dosimetry Activities in Ukraine // Reactor Dosimetry: Radiation Metrology and Assessment, ASTM STP 1398 (Proc. 10-th Intern. Symp. on Reactor Dosimetry, Osaka, Japan, 12 - 17 Sept. 1999.) - ASTM, West Conshohocken, PA, 2001. - P. 61 - 68.
2. *Grytsenko O.V., Vasylyeva E.G., Bukanov V.N. et al.* Improved surveillance program realizing at the WWER-1000 reactor of Rovno NPP Unit 4 // 12-th Intern. Symp. on Reactor Dosimetry, Gatlinburg, TN, USA, 8 - 13 May 2005, Programme & Book of Abstract. - Gatlinburg, TN, USA, 2005. - P. A08.


## ДОЗИМЕТРИЯ ОБРАЗЦОВ-СВИДЕТЕЛЕЙ, ОБЛУЧАЮЩИХСЯ В РЕАКТОРЕ ВВЭР-1000


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Изложены основные принципы работы пакета программ MCSS, являющегося основной частью методики определения функционалов нейтронного потока на образцы-свидетели, облучающиеся в реакторе ВВЭР-1000, разработанной специалистами ИЯИ НАН Украины. Кратко описана модернизированная программа образцов-свидетелей, которая реализуется в настоящее время на реакторе ВВЭР-1000 энергоблока № 4 Ровенской АЭС.


## ДОЗИМЕТРІЯ ЗРАЗКІВ-СВІДКІВ, ЩО ОПРОМІНЮЮТЬСЯ В РЕАКТОРІ ВВЕР-1000


**О. В. Гриценко, В. М. Буканов, В. Л. Дємьохін, О. Г. Васильєва, Ю. П. Малерс**



Викладено основні принципи роботи пакета програм MCSS, що є основною частиною методики визначення функціоналів нейтронного потоку на зразки-свідки, що опромінюються в реакторі ВВЕР-1000, розробленої спеціалістами ІЯД НАН України. Коротко описано модернізовану програму зразків-свідків, що реалізується в теперішній час на реакторі ВВЕР-1000 енергоблока № 4 Рівненської АЕС.