WWER-1000 Core and Reflector Parameters Investigation in the LR-0 Reactor

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Abstract

Measurements and calculations carried out in the core and reflector of WWER-1000 mock-up are discussed:

– the determination of the pin-to-pin power distribution in the core by means of gamma-scanning of fuel pins and pin-to-pin calculations with Monte Carlo code MCU-REA and diffusion codes MOBY-DICK (with WIMS-D4 cell constants preparation) and RADAR,

– the fast neutron spectra measurements by proton recoil method inside the experimental channel in the core and inside the channel in the baffle, and corresponding calculations in P_3S_8 approximation of discrete ordinates method with code DORT and BUGLE-96 library,

– the neutron spectra evaluations (adjustment) in the same channels in energy region 0.5 eV-18 MeV based on the activation and solid state track detectors measurements.

KEYWORDS: WWER-1000, mock-up, core, power distribution, neutron spectrum

1. Introduction

The zero power LR-0 reactor was designed for research of neutron parameters of the WWER type pressurized water reactors.

Suitable geometrical conditions and flexible technical arrangements of the LR-0 enable to construct full-scale physical models of an appropriate sector of the WWER type reactors in radial direction from the core to the biological shielding (included). The simulators of the reactor internals (core, baffle, and barrel) are located inside the LR-0 tank, the pressure vessel and biological shielding simulators - outside the tank.

The unique full-scale models (mock-ups) of WWER-440 and WWER-1000 reactors were developed in collaboration of NRI Řež, Škoda NM (Czech Republic) and RRC Kurchatov Institute (Russia) first of all for the researches in the pressure vessels and surveillance specimens dosimetry [1-4].

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The core of WWER-1000 mock-up (see Fig.1) is composed of 24 fuel assemblies (FA) with 2% ^{235}U enrichment, 6 FA with 3% ^{235}U enrichment, and 2 FA with 3.3% ^{235}U enrichment. The axial core dimension is equal to 125 cm. The dry channel for neutron spectrum measurement is positioned in the core. Control clusters are installed in symmetry positions and each control cluster consists of 3 absorbing pins (boron carbide); the typical position during measurements is one half of active length inserted. Critical concentration of boron acid is equal to 4.6 ± 0.1 g/l at moderator level 25 cm above core.

The measurements and calculations of fast neutron spectra were carried out in all representative points of mock-up (from barrel to biological shielding), important for the pressure vessel and surveillance specimens dosimetry benchmarking.

Detail measurements and calculations were carried out also in the core and reflector of mock-up and these mesurements and calculations can provide data for validation and improvement of WWER-1000 core calculation methods and codes.

Three types of mesurements and calculations are considered in the report:

– the determination of the pin-to-pin power distribution in the core,

– the fast neutron spectra measurements by proton recoil method inside the experimental channel in the core and inside the channel in the baffle,

– the neutron spectra evaluations based on the activation and solid state track detectors measurements in the same channels.

2. Core Power Distribution

The mock-up core power (fission rate) distribution is necessary to determine a source for neutron transport calculation. Data obtained on LR-0 WWER-1000 mock-up could be used also for core calculations benchmarking.

The experimental core fission rate distribution was obtained by means of gamma-scanning of fuel pins as an average of ^{140}La single peak (1596 keV) measurements and wide energy range (approximately 600 – 900 keV) measurements. Altogether 260 fuel pins were scanned with relative experimental uncertainties from 2 to 5 %. The measurements were arranged for scanning of 5 cm length of pin, the centre of the measured part was 62 cm above the bottom of the fuel (active part of pin).

The fission rate was measured in symmetrical positions (with respect to mock-up axis of symmetry – see Fig. 1) with two above mentioned independent measuring devices in each position. The obtained four measured values were averaged (as independent measurements) and measured uncertainties were evaluated.

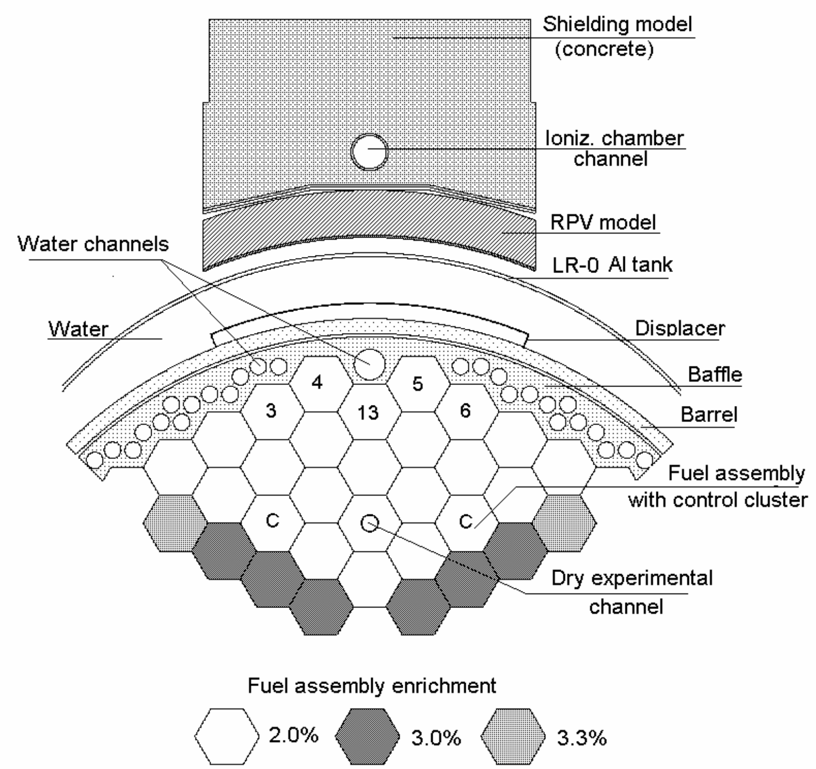
Pin-to-pin calculations of the WWER-1000 mock-up core power distribution were performed with several codes: Russian Monte Carlo code MCU-REA [5], Czech diffusion code MOBY-DICK [6] and Russian diffusion code RADAR [7].

The 3D calculation model of mock-up in 180^{o} sector was used in Monte Carlo calculations, and fission rate was calculated for all measured pins.

At the boundary of core, in the region very important for pressure vessel exposure evaluation, the neutron flux density is five times less than in the centre of the core. Due to this ratio the Monte Carlo calculation statistical uncertainty in the boundary region is about 2.5 times higher than in central FA. It was desirable to decrease the relative statistical uncertainty to 1–2 % in the FA 4, 5, and 13 near the central channel in baffle. Thus the calculations were done in two stages. The core was calculated with $50 \cdot 10^{6}$ histories of neutrons, simultaneously the neutron sources at the surfaces of the mentioned FA were accumulated and calculations inside these FA were continued starting from accumulated sources. After splitting of neutron sources the statistics was increased and the statistical uncertainty was decreased 2.5 – 3 times.

The calculations were performed with high number of neutrons in package - up to 2000 (200 neutrons in standard cases). The point-wise, sub-group and group structure of cross sections energy dependence was used in different neutron energy regions.

Figure 1: WWER-1000 mock-up horizontal section
(3, 4, 13, 5, and 6 – fuel assemblies mentioned in the text)



The basic factor contributing to the unreliability of the calculations is unsatisfactory knowledge of the gaps between FA and baffle. The uncertainty in the gap dimensions can be 2 – 3 mm which corresponds up to the 15 % uncertainty in power distribution of the boundary pin row (special evaluations of fission rate distribution sensitivity to gap dimension were done in separate model calculations). The core Monte Carlo calculations were done for 0 and 2 mm gaps. The differences between Monte Carlo calculations and experimental data are presented in the Fig. 2 for 3 outermost pin rows of FA 3,4,13,5,and 6 (row 1 is nearest to the baffle and includes pins located along all borders of assembly; rows 2 and 3 are parallel to row 1).

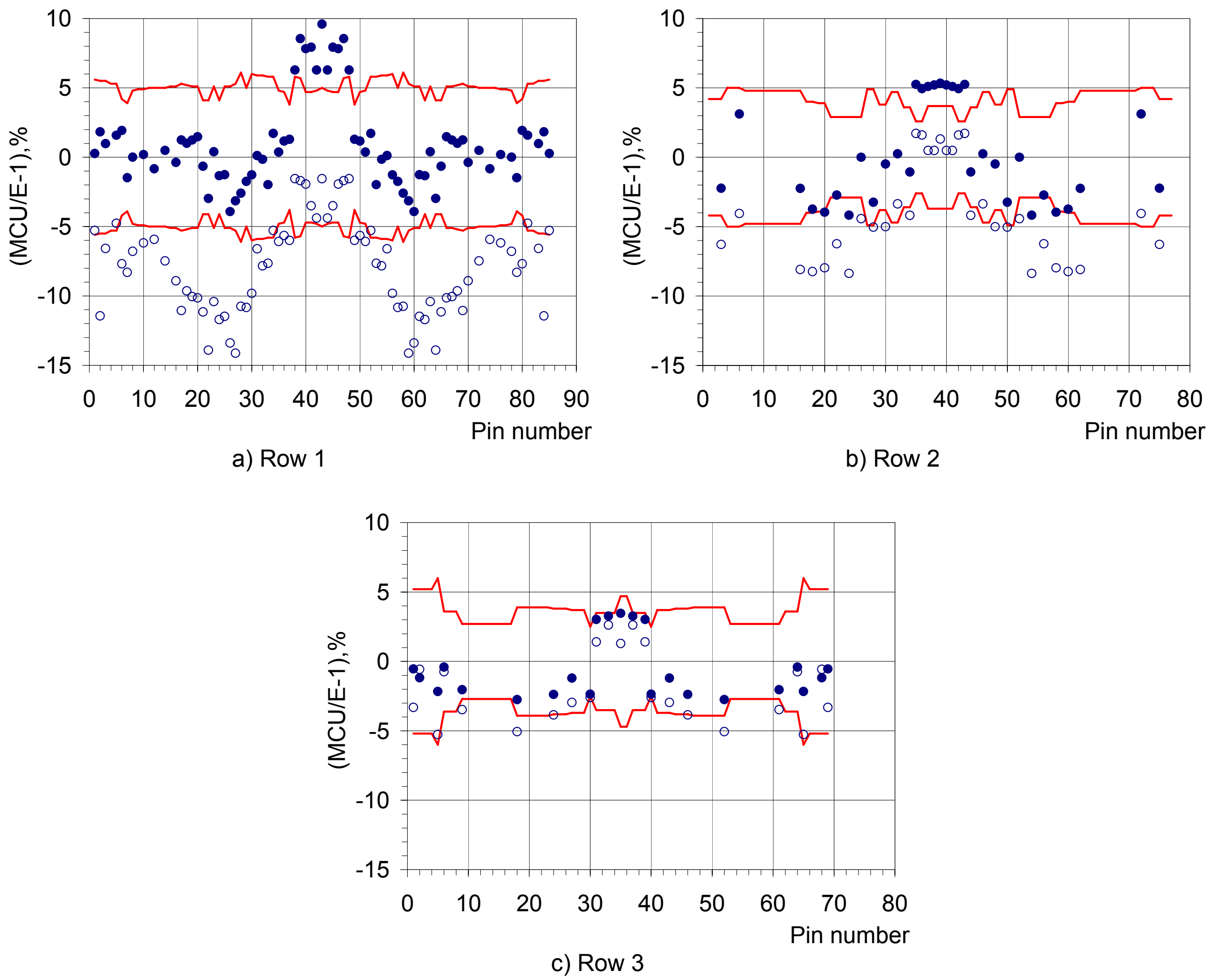
The results of calculations with MCU-REA agree with experiment in the range ± (4 – 5) % if the gap between FA No. 13 and baffle is 0 – 1 mm and about 2 mm for other FA. It corresponds to the known uncertainties in the mock-up geometry and should be taken into account at elaboration of model for neutron transport calculations.

Critical parameters and pin-to-pin power distribution were carried out also with diffusion code MOBY-DICK [6], which is used for WWER and mock-ups standard calculations in Czech Republic.

Core calculations were performed by this code in two-group finite difference approximation for 2D core model with an effective axial leakage ($H_{extr.}$ = 125+2*6.5 = 138 cm). Reflector was “covered” by grid of the same hexagons, so it is modeled as an extension of the core.

Two-group diffusion cross sections library for core and reflector (for mixture water-steel, for steel and pure water) was created for mock-up calculations using WIMS-D4 code.

Figure 2: Comparison of calculated (Monte Carlo) and measured relative power distribution in the first three outmost rows of fuel pins inside fuel assemblies 3, 4, 13, 5, and 6



●- gap 2 mm; ○- gap 0 mm; ___ combined error limits of experiment and MCU calculation

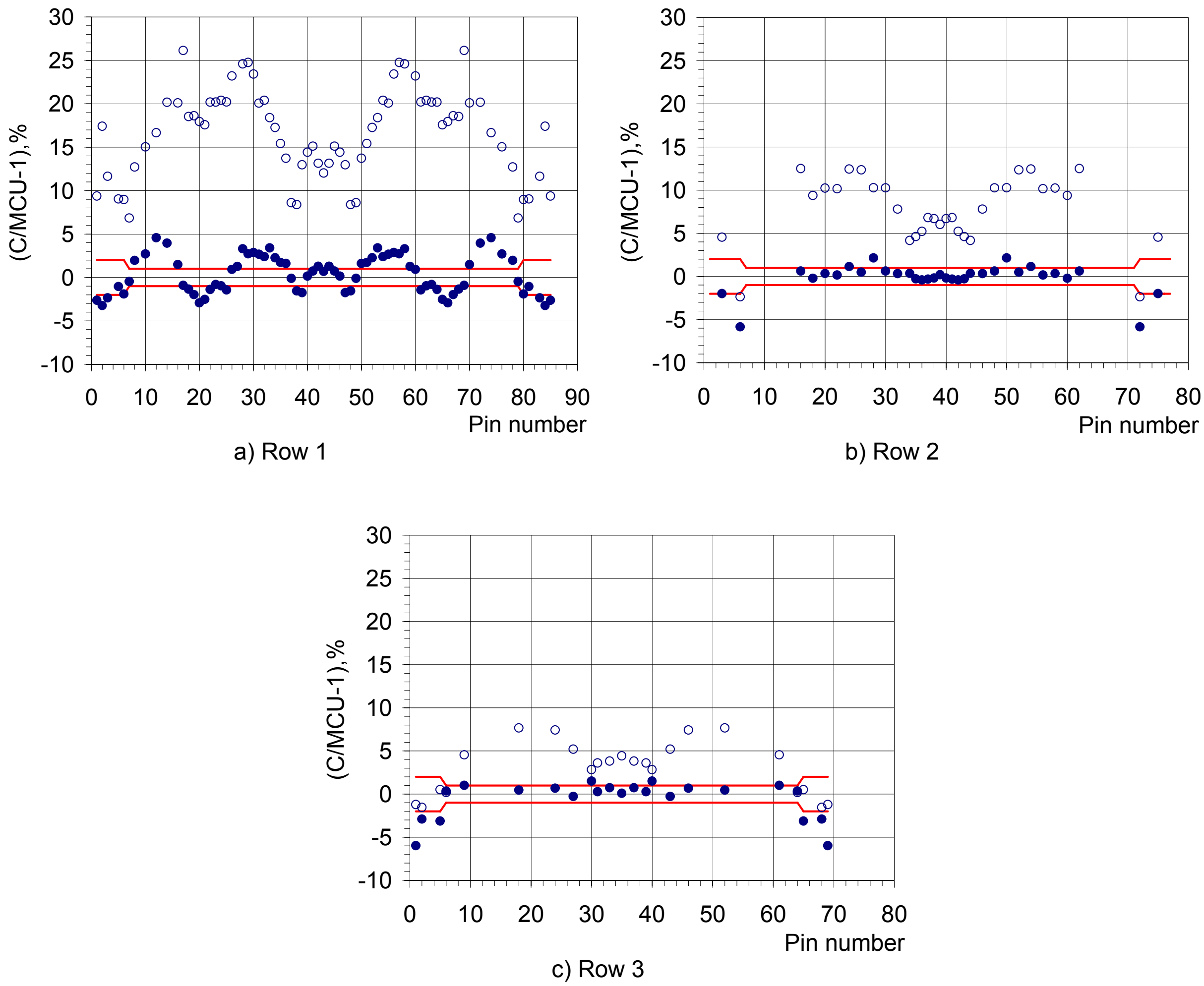
Russian code RADAR [7] contains two modules: for cell nuclear constants preparation and for spatial neutron field calculations.

Constant module calculates axial dependent multigroup neutron spectrum in FA and prepares the diffusion constants for cells. This module uses the same nuclear data library as MCU-REA code.

Spatial module provides the 3D diffusion multigroup pin-to-pin core calculation. Total number of energy groups can be from 1 to 63; in considered calculations 23 groups in fast and epithermal region and 9 groups in thermal region were used. Finite-difference schema realized in the spatial module is specified by introducing for every space mesh the nodal functionals prepared by constant module at spectral cells parameters calculations.

Results obtained with MOBY-DICK and RADAR are compared with MCU-REA calculations in the Fig.3. This comparison shows that more sophisticated 3D multigroup nodal code RADAR provides better description of boundary power distribution in comparison with 2D two-group code MOBY-DICK, so RADAR is preferable for determination of source for neutron transport calculations in mock-ups.

Figure 3: Comparison of calculated (MOBY-DICK, RADAR, and MCU) relative power distribution in the first three outmost rows of fuel pins inside fuel assemblies 3, 4, 13, 5, and 6



● RADAR; ○ MOBY-DICK; — statistical error limits of MCU calculation

3. Neutron Spectrum

Fast neutron spectra were measured by proton recoil spectrometer with two cylindrical stilbene scintillators (1×1 and 2×2 cm) [8] and the set of 3 spherical proportional counters (diameter 4 cm) with different hydrogen gas filling (100, 400, 1000 kPa) [9]. The pulse shape neutron-gamma discrimination was used for scintillator measurements and for chamber measurements in energy region below 100 keV approximately.

The response functions of spectrometer were proved in monoenergetic neutron beams [10]. The spectrometer used to be checked before measurements in known reference fields [11].

The power monitoring system plays an important role, if a few measurements with different detectors are combined to evaluate the neutron spectrum. Several fission chambers of different efficiency in different points of LR-0 serve as the power monitoring system, which was independent of the LR-0 operational and safety monitoring system. The evaluated long-term stability of this system is equal to 5%. The actual uncertainty within a couple of days is ~ 2%.

The activation and fission track measurements were performed in the same positions as proton recoil spectrometry in the core and reflector. Due to LR-0 power restrictions (no more 5 kW for one hour) the measurements were carried out at low levels of neutron flux and fluence. These circumstances influenced the list of chosen activation and track detectors. The used detectors are listed in the Table1.

Table 1: Activation and track detectors for neutron spectrometry

Thermal and resonance neutrons			Fast neutrons		
Material	Reaction	Sensitivity region	Material	Reaction	Sensitivity region, MeV
Metal (nat.)	$^{45}Sc(n,\gamma)$	Therm. neutrons	NpO_2	$^{237}Np(n,f)$	>0.55
6% alloy (Al)	$^{176}Lu(n,\gamma)$	Therm. neutrons	Metal (natural)	$^{115}In(n,n')$	>1.0
UO_2	$^{235}U(n,f)$	Therm. and res. n.	UO_2	$^{238}U(n,f)$	>1.5
Pu oxides	$^{239}Pu(n,f)$	Res. neutrons	Metal (natural)	$^{47}Ti(n,x)$	>2.2
Alloy with Al	$^{115}In(n,\gamma)$	Therm.and res. n.	Metal (natural)	$^{58}Ni(n,p)$	>2.5
1% alloy (Al)	$^{197}Au(n,\gamma)$	Therm. and res. n.	Natural	$^{32}S(n,p)$	>3.0
Metal (nat.)	$^{93}Nb(n,\gamma)$	Therm. and res. n.	Metal (enr. 96%)	$^{54}Fe(n,p)$	>3.0
1% alloy (Al)	$^{139}La(n,\gamma)$	Therm. and res. n.	Metal (natural)	$^{56}Fe(n,p)$	>6.3
1% alloy (Al)	$^{59}Co(n,\gamma)$	Therm. and res. n.	Metal (natural)	$^{46}Ti(n,x)$	>4.6
3% alloy (Al)	$^{63}Cu(n,\gamma)$	Therm. and res. n.	Metal (enr. 99%)	$^{63}Cu(n,\alpha)$	>6.0
ceramics (60% Na)	$^{23}Na(n,\gamma)$	Res. neutrons	Metal (natural)	$^{24}Mg(n,p)$	>7.0
			Metal (natural)	$^{27}Al(n,\alpha)$	>7.3
			Metal (natural)	$^{93}Nb(n,2n)$	>10

The adjustment of neutron spectra in energy region 0.5 eV-18 MeV was done using results of activation and track measurements.

Thermal and epithermal parts of spectrum were described by Maxwell, Fermi and transient distributions. The measured reaction rates were used for determining the spectra parameters in the core channel. Neutron temperature was determined using the temperature sensitive reaction $^{176}Lu(n,\gamma)$. Neutron flux density in the baffle channel is significantly lower than in core channel, and less number of activation reactions could be used (in particular reactions $^{139}La(n,\gamma)$ and $^{176}Lu(n,\gamma)$ were not used). Neutron spectra parameters in baffle channel were determined using spectrum in the core channel as initial approximation.

The measured reaction rates are presented in the Table 2. Reaction rates were calculated using adjusted spectra and calculation to measured ratios ("Adj/Exp") are shown in this Table also.

Thermal neutron flux density was obtained (4.07E+12 n/m^{2}s)±3.5% in the core and (1.08E+12 n/m^{2}s)±14% in the baffle. Comparison of measured and calculated reaction rates illustrates good consistency of input data and demonstrates the reliability of thermal and epithermal spectrum parameters determination.

Table 2: Measured and calculated reaction rates comparison in thermal and resonance neutrons region

Reaction	Core				Baffle			
	Without Cd cover		With Cd cover		Without Cd cover		With Cd cover	
	RR, 1/s Experim.	Adj/Exp	RR, 1/s Experim.	Adj/Exp	RR, 1/s Experim.	Adj/Exp	RR, 1/s Experim.	Adj/Exp
^{176}Lu(n,γ)	2.00E-12	1.010	7.65E-14	0.986				
^{45}Sc(n,g)			9.42E-16	0.994	3.40E-15	1.002	5.88E-17	1.013
^{115}In(n,γ)	2.84E-13	0.998	1.91E-13	1.000	3.95E-14	0.931		
^{197}Au(n,γ)	1.63E-13	1.005	1.26E-13	1.001	2.11E-14	1.000	7.24E-15	0.995
^{93}Nb(n,γ)	1.20E-15	1.007	9.89E-16	1.038				
^{139}La(n,γ)	4.72E-15	0.993	1.04E-15	1.005				
^{59}Co(n,γ)	1.76E-14	1.011	6.45E-15	1.017	4.72E-15	1.029	2.17E-16	1.029
^{63}Cu(n,γ)	2.32E-15	0.977	4.95E-16	0.956	6.79E-16	1.004	2.47E-17	1.011
^{23}Na(n,γ)	2.46E-16	0.993	2.63E-17	1.025	7.14E-17	0.989	1.66E-18	0.977
^{239}Pu(n,f)			2.80E-14	0.941				

Fast neutron spectra were adjusted with code MIXER (directional discrepancy method) [12] using measured reaction rates and calculated spectra as initial approximation. Calculations were done in P_3S_8 approximation of discrete ordinates method with code DORT [13] and BUGLE-96 library [14] and calculated spectra were transformed in SAND-2 format for adjustment.

Comparison of measured and calculated reaction rates (normalized to 58Ni(n,p) reaction rate) is presented in the Table 3. In this Table "Adj", "Calc", and "Calc(ES)" means reaction rates calculated using adjusted spectrum, spectrum calculated with DORT code and spectrum measured by proton recoil method respectively.

Table 3: Measured and calculated reaction rates comparison in the fast neutrons region

Reaction	Core				Baffle			
	Experim.	Adj/Exp	Calc/Exp	Calc(ES)/ Exp	Experim.	Adj/Exp	Calc/Exp	Calc(ES)/ Exp
^{237}Np(n,f)	1.68E+01	1.01	1.042	1.002	1.69E+01	1.00	0.990	0.980
^{115}In(n,n')	2.10E+01	1.01	1.024	1.014	2.10E+01	0.972	0.920	0.910
^{238}U(n,f)					3.30E+01	0.968	0.910	0.970
^{58}Ni(n,p)	1.000	0.99	1.100	1.160	1.000	0.991	1.090	1.070
^{54}Fe(n,p)	7.56E-01	0.99	1.110	1.170	7.66E-01	0.981	0.950	0.940
^{46}Ti(n,x)	9.63E-02	0.94	1.130					
^{63}Cu(n,α)	4.08E-03	1.04	1.660					
^{93}Nb(n,2n)	4.63E-03	1.00	1.150					

The absolute value of ^{58}Ni(n,p) reaction rate is equal to (8.36±0.18)E-17 s^{-1} in the core and (3.32±0.10)E-18 s^{-1} in the baffle. The significant discrepancies for high threshold reactions are in the core.

Spectral indices (ratios of integral neutron flux densities above different thresholds) are compared in the Table 4. In this Table "Adjusted" SI obtained from the adjusted spectra, "Measured" – from the spectra measured by proton recoil method, "Calculated" – from spectra calculated by discrete ordinates method with code DORT and BUGLE-96 nuclear data library.

In all cases the “Adjusted” SI are higher than “Measured” ones, and especially significant discrepancy is in the core. Calculation underestimates the “Adjusted” SI in all cases, and overestimates “Measured” ones in the core. Agreement between calculation and measurement in the baffle is good. Probably activation measurements as well as measurements by proton recoil method should be repeated in the core, and the possible perturbation effects should be estimated.

Table 4: Measured and calculated spectral indices

Spectral index	Point	Adjusted	Measured	Calculated	A/M	C/A	C/M
SI(0.5/3.0)	Core	5.62	4.78	5.31	1.18	0.94	1.11
SI(0.5/3.0)	Baffle	5.57	5.32	5.22	1.05	0.94	0.98
SI(1.0/3.0)	Core	4.10	3.41	3.67	1.20	0.90	1.08
SI(1.0/3.0)	Baffle	3.98	3.67	3.68	1.08	0.92	1.00

4. Conclusions

The complex investigations on WWER-1000 mock-up (power distribution, activation and proton recoil spectra measurements) clearly demonstrate that independent experiments and calculations of different types are necessary for data reliability and self consistency improvement and carrying out of benchmark.

Acknowledgements

Described investigations were carried out under TACIS SRR-2/95 project, IAEA RER 4/017 technical cooperation project, and REDOS project of 5th Framework Program of European Community.

Authors express thanks to their colleagues E.B. Brodkin, A.L. Egorov, V.I. Vikhrov, F. Hudec, S. Pošta, B. Jansky, V. Krysl, T.S. Zaritskaya and reactor LR-0 staff for valuable contribution to the described investigations.

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