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CALCULATION OF NEUTRON FLUXES IN BIOLOGICAL SHIELD OF THE TRIGA MARK II REACTOR

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ABSTRACT

The complete calculation of neutron fluxes in biological shield and verification with experimental results is presented. Calculated results are obtained with TORT code (TORT-Three Dimensional Oak Ridge Discrete Ordinates Neutron/Photon Transport Code). Experimental results used for comparison are available from irradiation experiment with selected type of concrete and other materials in irradiation channel 4 in TRIGA Mark II reactor. These experimental results were used as a benchmark.
Homogeneos type of problem (without inserted irradiation channel) and problem with asymmetry (inserted beam port 4, filled with different materials) were of interest for neutron flux calculation. Deviation from material data set up as original parameters is also considered (first of all presence of water in concrete and density of concrete) for type of concrete in biological shield and for selected type of concrete in irradiation channel. BUGLE-96 (47 neutron energy groups) library is used. Excellent agreement between calculated and experimental results for reaction rate is received.

1 INTRODUCTION

Determination of the residual activity for reactor structure materials is one of the main tasks to solve after the reactor lifetime. One of the most important activated part (reactor structure material) is the biological shield (concrete). Generally, activation depends on the material properties and neutron flux. Since we know element structure for materials of interest, the only missing data is neutron flux in materials used as reactor structure materials, respectively. Neutron flux can be obtained either with experimental determination or using calculation methods. Experimental neutron activation data must be measured due to verification of the results from calculations.
Calculation of neutron fluxes in biological shield (concrete) of the TRIGA Mark II research reactor in Ljubljana is performed with computer code TORT [2] and BUGLE-96 library [3] in

four steps. RΘZ geometry is used for model. Determination of model (final mesh, proper choosing of the S_n and P_n, final geometrical symmetry, boundary conditions, …), establish of the cross section library, element structure [1], [5] for homogeneous type of whole reactor body was the first step. Implant of model for port number 4 in the whole reactor body and element structure [1], [5] for materials filled in port number 4 with new cross section library was next step. The third step was varying of density and water content in ordinary concrete and barytes concrete for whole reactor body with port number 4. The last step was comparison between calculation and measured data (reaction rates) from experiments [4] for data form third step. Very good agreement between calculated and experimental results for reaction rate is received.

2 GEOMETRY AND CALCULATED MODEL DESCRIPTION

Geometry description can be divided in two parts:

- Homogeneous type
- Homogeneous type with inserted beam port 4: non-homogeneous type

Both models are created in RΘZ geometry. Basic calculated model mesh for non-homogeneous type is the same as in the homogeneous case except for additional cells (only in Θ direction, r and z direction descriptions remain the same structure) describing beam port 4. Material description was also the same [1], [5] for homogeneous core was used fuel assembly arrangement as in core number 169 (54 fuel elements) [1]. Final library (obtained from BUGLE-96 [3], 47 neutron energy groups) has 40 nuclides in 16 mixtures. Upscattering [3] was not used in calculations. The same library was used for both geometry types.

2.1 Geometry description for homogeneous type

The homogeneous type of geometry is the basis for establishing calculated mesh for both for both geometry types. Heterogeneous core is homogenized. All structural materials of the heterogeneous core are volume weighted. So, one can determine final atomic densities for homogeneous core used in library description, according to volume shares of the element structure. Detailed material description used in both homogeneous and non-homogeneous types is presented in Table 1. This table is a foundation for determination of the element composition for homogeneous core with volume weighting (geometrical description needed for material - volume determination) and for material description in beam port 4 for final library (non-homogeneous geometry type). Homogeneous geometry type represents the basis for analysis of the spatial and energy neutron flux distribution inside whole reactor body. Figures 1 and 2 describe the geometry used for homogeneous type.

2.2 Homogeneous type with inserted beam port 4: Non-Homogeneous type

Homogeneous geometry type is used as the rudimentary groundwork for determination of the non-homogeneous type. The only difference is in inserting of additional mesh spacing in Θ-direction in only vertical slides within beam port 4 (r-direction and z-direction mesh divisions remain unchanged). Whole length of beam type 4 is filled with different materials. Detailed description is in Figure 3. Each material part is divided on various number of volume bodies ike in the Figure 4 and division number depends on the specific material lenght to preserve material volume in the best way.

Table 1: Materials used in both calculated models

Material	Density	Component	Content	Atom Density
	(g/cm^3)		(wt. %)	atoms/barn-cm
Fuel mixture U-ZrH (*)	6,04495	H	1,5298	5,52530E-02
fuel elements		Zr	86,5302	3,45300E-02
		U238	9,5639	1,46250E-03
		U235	2,3761	3,68010E-04
Fuel mixture U-ZrH (*)	6,15776	H	1,5298	5,62840E-02
Control rods		Zr	86,5302	3,51750E-02
		U238	9,5639	1,48980E-03
		U235	2,3761	3,74870E-04
Zr rod (*)	6,49	Zr	100	4,28430E-02
Absorber (B_4C) (*)	2,48	B10	15,522	2,14430E-02
(Boral)		B11	62,478	8,63100E-02
		C	22	2,73550E-02
Stainless steel - SS304 (*)	7,889	Fe	66,84	5,68600E-02
		Cr	19	1,73600E-02
		Ni	10	8,09480E-03
		Mn	2	1,72950E-03
		Si	2	3,38310E-03
		C	0,08	3,16430E-04
		P31	0,04	6,13530E-05
		S	0,04	5,92560E-05
Aluminum (*)	2,7	Al	100	6,02620E-02
Supporting disc (*)	10,2	Mo	100	6,40250E-02
in fuel elements				
Graphite (*)	1,6	C	100	8,02210E-02
Water - Core Coolant (*)	0,9975	H	11,19	6,66890E-02
(T=23^{0}C)		O	88,81	3,33440E-02
Void regions (air) (*)	0,0013	N14	77,79	4,34790E-05
		O	22,21	1,08680E-05
Concrete - Type 04	2,25	H	0,67	9,41046E-03
		C	7,92	8,91207E-03
		O	49,15	4,16670E-02
		Na	0,00	0,00000E+00
		Mg	0,22	1,11497E-04
		Al	0,49	2,51093E-04
		Si	7,27	3,52185E-03
		K	0,11	3,46555E-05
		Ca	33,63	1,13596E-02
		Fe	0,29	7,27867E-05
		S	0,19	8,45009E-05
Lead	11,34	Pb207		3,29580E-02
Wood		H		2,37700E-02
		C		1,42600E-02
		O		1,18900E-02
Barytes	3,7	H	0,57	1,32643E-02
Heavy Concrete		O	32,42	4,51223E-02
		Al	0,33	3,30327E-04
		Si	0,97	7,93355E-04
		Ca	4,21	2,33503E-03
		Fe	0,59	2,39387E-04
		S	11,35	7,92055E-03
		Ba138	48,10	7,80439E-03
		Mg	0,71	1,13577E-04
		C	0,61	9,73520E-05

Materials marked with ' (*) ' in Table 1 are used in core homogenization. Modeled beam port 4 has a shape like ‘christmas tree’.

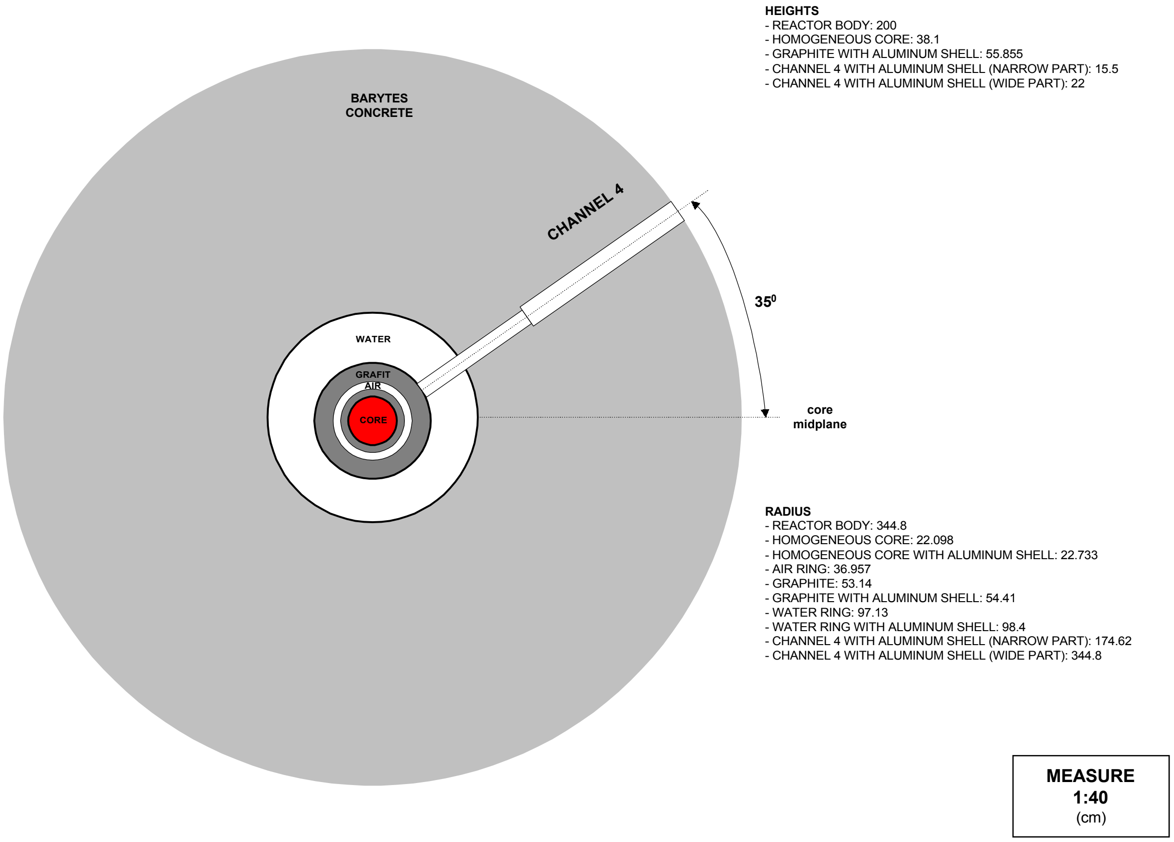

Figure 1: Schematic top view of the TRIGA Mark II research reactor model with beam port 4

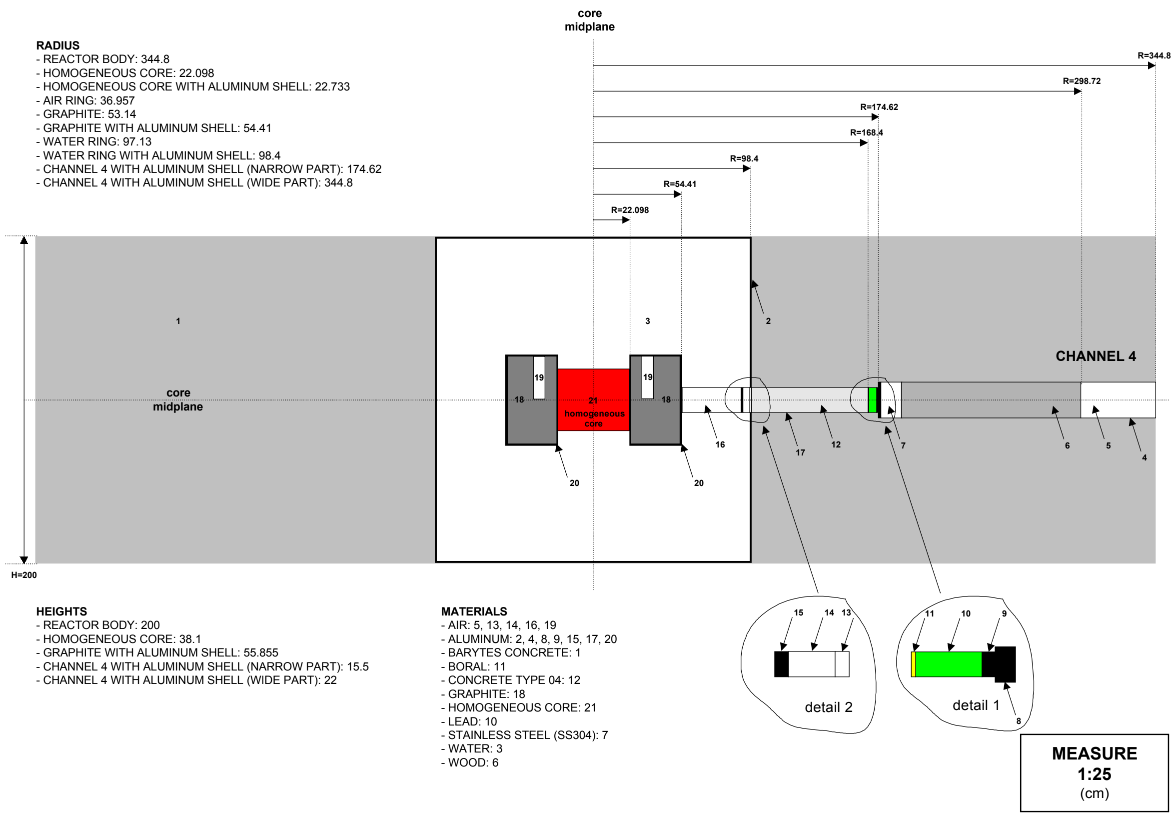

Figure 2: Schematic side view of the TRIGA Mark research reactor model with beam port 4

Mesh size increased from 63936 cells (homogeneous type) to 177600 cells (non-homogeneous type) for filled beam port 4 with various materials over the length as presented on Figure 3.

With inserted beam port 4 in the homogeneous reactor body it is possible to:

- Make a comparison between calculated an measured data (reaction rate for irradiated materials)
- Estimate the influence of the horizontal irradiation channels (beam port 4) on the distribution of the neutron flux in biological shield

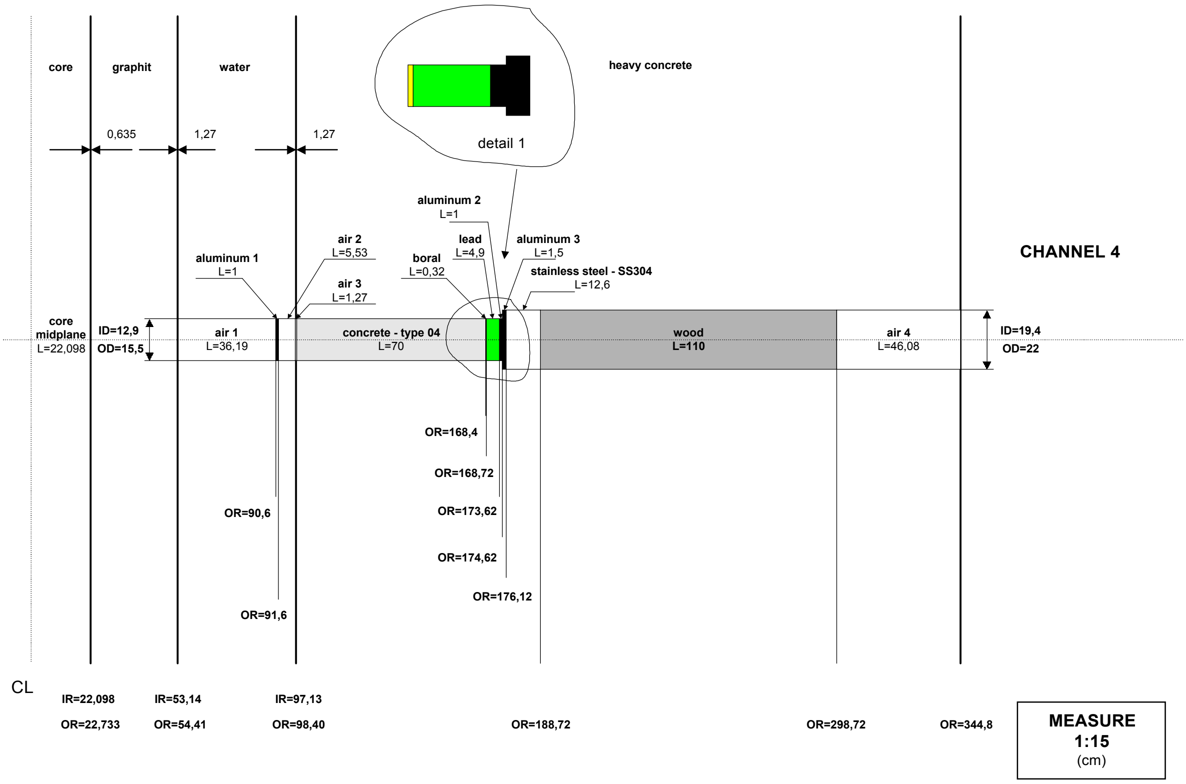

Figure 3: Beam port 4 filled with different materials during experiment

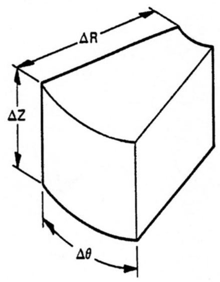

Figure 4: Spatial cell used in modeling beam port 4 - ‘pie slice’, used generally in RΘZ

Sample holder is inside the beam port 4. It is lying with angle 35^0 regarding to core midplane.

3 DEVELOPMENT AND VERIFICATION OF THE CALCULATED MODEL

The basis for obtaining a good non-homogeneous model (used for comparison with measured data) is verified homogenous model. There is a huge amount of questions arise from considering such complicated task.

3.1 Computer code TORT brief description

Computer code TORT [2] is used for generating neutron fluxes in whole reactor body. The principal application is to deep penetration of neutron and photons. TORT calculates the flux or fluence of neutrons and/or photons throughout 3-D and 2-D systems due to particles incident upon the system's external boundaries, due to fixed sources, or due to sources generated by interaction with the system materials. The transport process is represented by the Boltzmann transport equation. The method of discrete ordinates is used to treat the directional variable, and a multigroup formulation treats the energy dependence. Anisotropic scattering is treated using a Legendre expansion. Various methods are used to treat spatial dependence, including nodal and characteristic procedures. Reactor eigenvalue problem can also be solved. Either cylindrical (RΘZ) or cartesian (XYZ) geometry is supported.

3.2 Presentation of the principal objects about solving homogeneous case with TORT

Boundary conditions need to be proper. TORT input model (RΘZ geometry) used (our geometry description) in all cases reflected boundary condition for R-directions (both sides), Z-direction (both sides) and periodic boundary condition for Θ-direction (both sides).
P_3 option (scattering expansion) is used. Model is not tested regarding to various P_n choice.
S_{10} (number of directions, for n=10, 140 directions) option is used in all cases. This S_n option is also tested to establish neutron flux dependence from different S_n choice with P_3 option and reference mesh 13 (63936 cells). Results are presented on Figure 6.
Library is created completely by user. Data for creation is received from BUGLE-96 [3] library (47 energy neutron groups) set and relevant problem. Because geometrical model describes all reactor body with different material compositions along the radius the user must be aware about correct chose of microscopic cross section set for each element in pertinent material composition due to different neutron spectra used for generating microscopic cross section set on general reactor material structure (also consider P_n choice for library description). Mixtures with element compositions (atomic densities) must be included in the library along with several factors necessary. Computer code GIP [2] is used to create applicable library data (from input library) used in TORT code. GIP code starts before TORT code execution. Library BUGLE-96 contains also set of response cross sections for different reactions.

Table 2: Different mesh sizes used for reference mesh determination (mesh 13)

calc. num.:	mesh number	number of cells	R-axis	Θ-axis	Z-axis
			divisions	divisions	divisions
1	MESH 3	4050	27	5	30
2	MESH 4	8820	35	7	36
3	MESH 2	19656	42	13	36
4	MESH 5	33696	52	18	36
5	MESH 8	43776	64	18	38
6	MESH 11	53568	62	24	36
7	MESH 12	55296	64	24	36
8	**MESH 13 (reference - homogeneous case)**	**63936**	**74**	**24**	**36**
9	**MESH 15 (non-homogeneous case)**	**177600**	**74**	**24 (26 axial layers)**	**36**
	Θ-direction (double division)			**152 (10 axial layers)**	

The homogeneous model is also tested to different angle symmetries (90^0, 180^0, 360^0). The problem is symmetrical, so 90^0 (proven choice) is used as homogeneous model. Mesh size is decreased in great amount. The remaining number of calculated cells from reduced mesh size is applied in beam port 4 mesh establishment in non-homogeneous case.

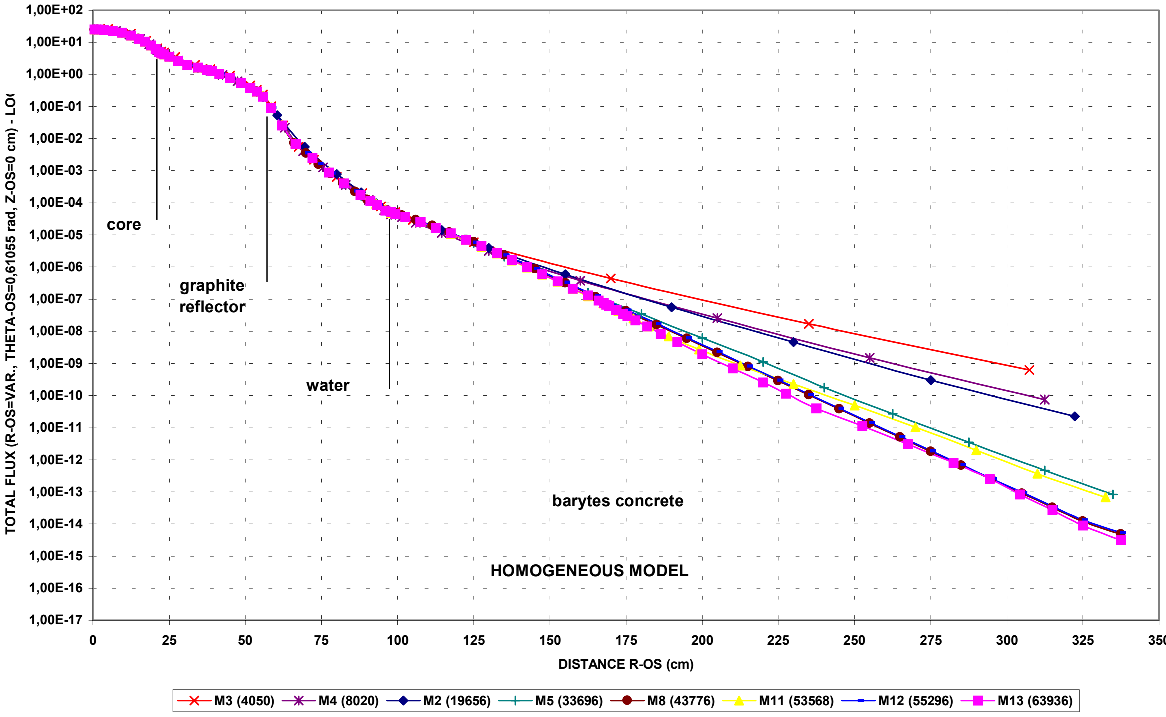

Figure 5: Different mesh sizes calculated with $S_{10}P_3$ (see Table 2)

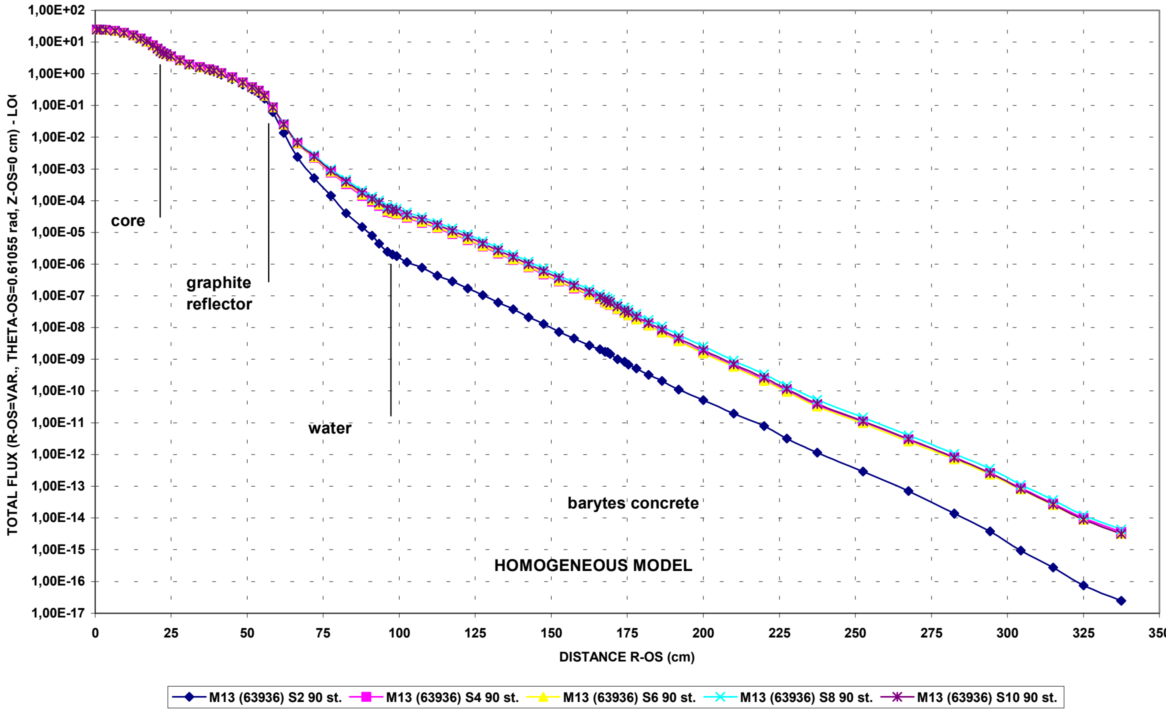

Figure 6: Mesh 13 (63936 cells) calculated with different S_n and P_3

Mesh size is tested in very extensive way. Mesh 13 is reference mesh size for homogeneous model, obtained after convergence mesh size process (Figure 5). Neutron fluxes for various mesh sizes are on Figure 5. Mesh size compositions used in Figure 5 are described in Table 2. S_2 (8 spatial directions) used as 'directional quadrature set' for calculation with mesh 13 discovers a great impact of anisotropic scattering on size of scalar flux (8, 32, 60, 96, 140 spatial directions for S_2, S_4, S_6, S_8, S_{10} option). S_2 'directional set' has lower scalar flux in the barytes concrete material structure and flux decreasing with increasing radius. Decreasing starts at the end of graphite material structure, about 54.5 cm far from vertical core midplane at the beginning barytes concrete. At the of the biological shield made from barytes concrete (about 340 cm far from vertical core midplane), scalar flux calculated with S2 'directional set' is lower for factor 1E-2 comparing with other S_n 'directional sets'.
S_{10} (with 140 spatial directions) is reference 'directional set' used for all cases (geometries used in this work).
All results obtained by TORT are relating to center points of the mesh cells from geometry point of view, 'diamond' system.
Fixed source energy spectrum is taken from procedure for creating library BUGLE-96 [3]. Source is uniformly distributed in homogeneous core volume.

4 RESULTS

Non-homogeneous geometry type of whole reactor body is described in chapter 2.2. Detailed material arrangement inside beam port 4 is presented on Figure 3.

4.1 Calculation of neutron fluxes in non-homogeneous type (with beam port 4)

Mesh division is preserved for R-direction and Z-direction from homogeneous case. The discontinuity is affected only in Θ-direction between homogeneous type and non-homogeneous type. Mesh 13 (63936 cells) is reference mesh for homogeneous type.

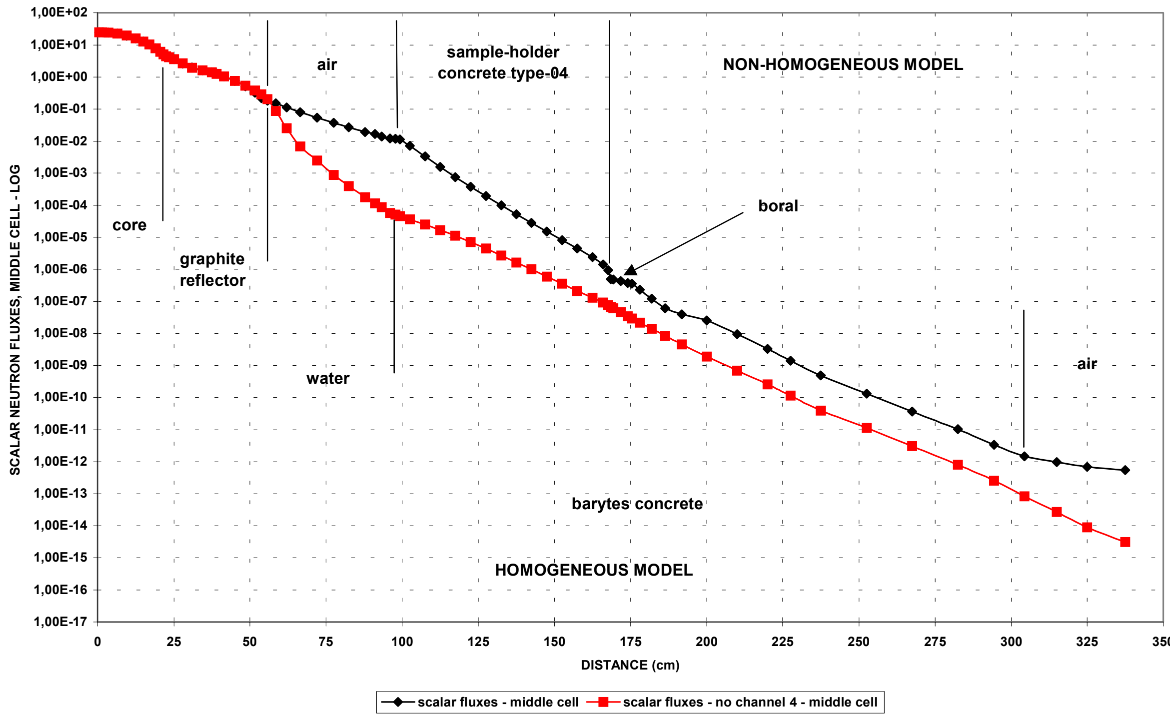

Figure 7: Scalar neutron fluxes in the centerline beam port 4 obtained with meshes 13 and 15

Mesh 15 (177600 cells) is reference model in non-homogeneous type (homogeneous type with inserted beam port 4 in whole reactor body). Mesh 15 has double division in Θ-direction. 90^0 symmetry is used in all cases. Scalar neutron flux in the centerline of the beam port 4 is on Figure 7 together with scalar neutron flux across the barytes concrete. Scalar neutron flux in barytes concrete is obtained with (mesh 13) calculation regarding to homogeneous case at the same angle 35^0 from core horizontal midplane as in case with beam port 4 with mesh15, non-homogeneous case. Slope is less inclined in the lenghts with air (first air length with radius from 54.41 cm to 90.6 cm and second part with radius from 298.72 cm to 344.8 cm across the centerline in the beam port 4) comparing to the scalar flux in homogeneous case. Scalar neutron flux drops with more intensity in water region (radius between 54.41 cm to 97.13 cm) than in barytes concrete in the homogeneous case (radius over the 98.4 cm to 344.8 cm). Spatial part in the beam port 4 filled with concrete type-04 (radius from 98.4 cm to 168.4 cm) has more rapidly decreasing scalar neutron flux than within the same radius interval in homogeneous cas with barytes concrete. Concrete type-04 has a bigger share of light nuclides than barytes concrete (Table 1) and greater possibility to moderate thermal neutrons. Boral at radius 168.72 cm is a reason for scalar neutron flux dropping in beam port 4.
Beam port 4 is settled with its centerline on horizontal core midplane. 10 axial layers divide in Z-directions beam port 4 with 152 angle sections in Θ-direction. The remaining 13 axial layers below and above beam port axial length have 24 angle sections in Θ-direction. Total number of axial layers is 36 in both meshes 13 and 15, as can be seen from Table 2.
Library is common for both meshes 13 and 15. It is created flexible enough to satisfy all compositions requirements for mesh types with different material arrangement and element material changes.
Figure 8 presents neutron spectrum normalized with lethargy width (47 energy neutron groups) in four mesh cells across the centerline of beam port 4. First two spectrums at radius 99.2 cm (beginning of the concrete type-04 sample) and 147.5 cm are in concrete type-04 material.

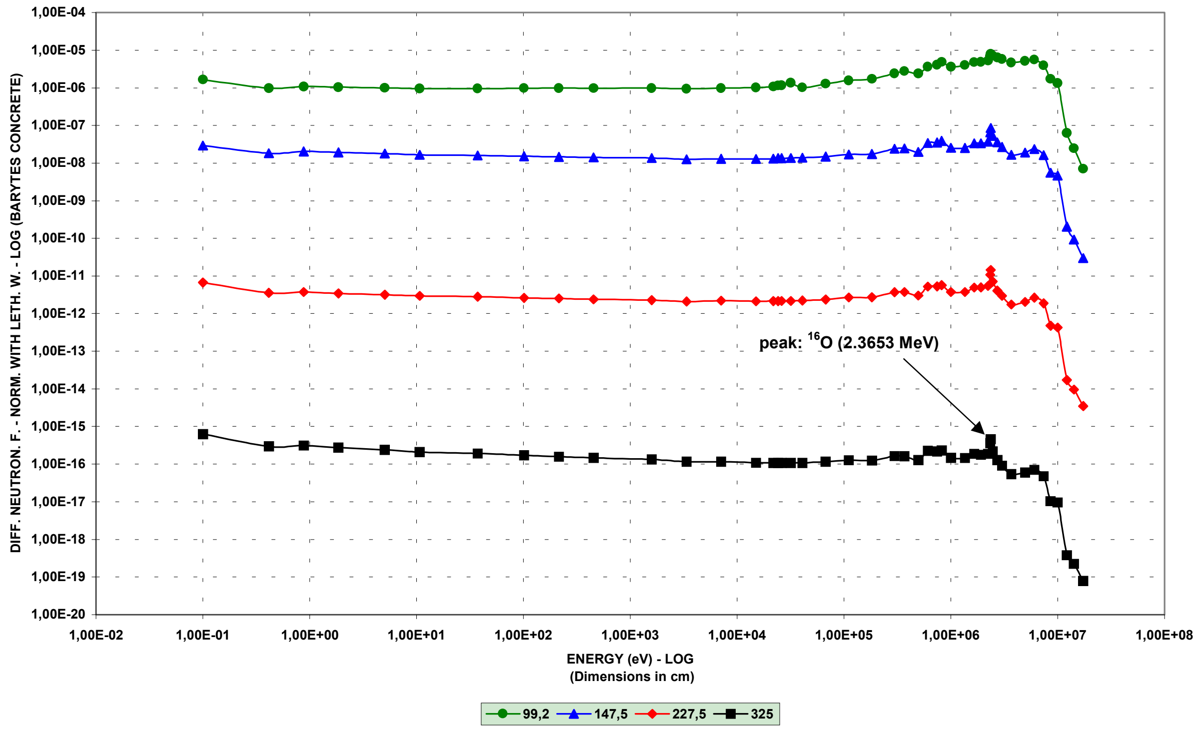

Figure 8: Neutron spectrum on different locations at the centerline inside beam port 4

Third calculated spectrum is at radius 227.5 cm in wood. The last spectrum is at radius 325 in the air part in beam port 4. All spectrums are decreasing in absolute manner like in Figure 7. Peak can be observed at energy point 2.3653*1E6 eV in the last three neutron spectrum curves on Figure 8. This is due to peak minimum in O^{16} microscopic cross section worth from BUGLE-96 library [3].

4.2 Comparison between measured and calculated data

Measured data is used from irradiation experiment with detailed description in reference [4]. One group of gold and nickel foils was irradiated for approximately one day at full reactor power at 250 kW with no polyethylene filter without interruptions. Second set of gold and nickel foils was irradiated with polyethylene filter in place (before concrete type-04 in beam port 4). Pure gold foils used in experiments were in form of small thin discs (8 mm in diameter and 50 μm thick). Gold wire was made of 0.1 % diluted gold in aluminum wire (1 mm diameter). Nickel wire was made of 99.982 % pure nickel in form of thin wire (0.75 mm in diameter). Samples are distributed within sample-holder (aluminum tube filled with concrete type-04 in beam port 4) on 8 different locations along whole reactor body radius. Detailed description is in reference [4].
Data of measured reaction rates for nuclear reaction $^{197}Au(n,\gamma)^{198}Au$ is used for verification of the calculated neutron fluxes with TORT code. Ratio of reaction rates at the beginning and at the end of sample-holder (60 cm distance in concrete type-04) is compared between measured and calculated data. Constant neutron flux Φ(E) and negligible target nuclide burnup (^{197}Au used in comparison, constant concentration N_0 of target nuclide during irradiation)) are adopted. With constant concentration target nuclide N_0, ratio of reaction rate is:

$$ratio = R = \frac{\int \sigma_{^{197}Au(n,\gamma)}(E)\,\Phi_0(E)\,dE}{\int \sigma_{^{197}Au(n,\gamma)}(E)\,\Phi_{60}(E)\,dE} \tag{1}$$

$\Phi_0(E)$ is neutron flux at the beginning of sample-holder (concrete type-04) and $\Phi_{60}(E)$ is neutron flux at the 60 cm deep in the concrete of sample-holder (concrete type-04). Library BUGLE-96 contains response function for nuclear reaction $^{197}Au(n,\gamma)^{198}Au$ (47 energy groups). Energy dependent neutron flux Φ(E) is calculated with TORT code for identical spatial points in sample-holder (concrete type-04) in non-homogeneous type model using different parameters described in Table 3.

Table 3: Impact of parameters changing on neutron flux in sample-holder (concrete type-04)

CONCRETE TYPE-04		**BARYTES CONCRETE**	
density = 2,25 g/cm^3	**water = 6 %**	**density = 3,7 g/cm^3**	**water = 5 %**
water – 4 %	density - 2 g/cm^3	water – 3 %	density - 3,6 g/cm^3
water – 6 % (reference)	density - 2,25/cm^3 (refernce)	water – 5 % (reference)	density - 3,7 g/cm^3 (reference)
water – 8 %	density - 2,5 g/cm^3	water – 7 %	density - 3,8 g/cm^3

Uncertainty is estimated due to errors in concrete composition used with changing concrete density and concrete water content for approximately ± 10% or more of the reference values (bolded values in Table 3) for density and water content for both types of concrete.

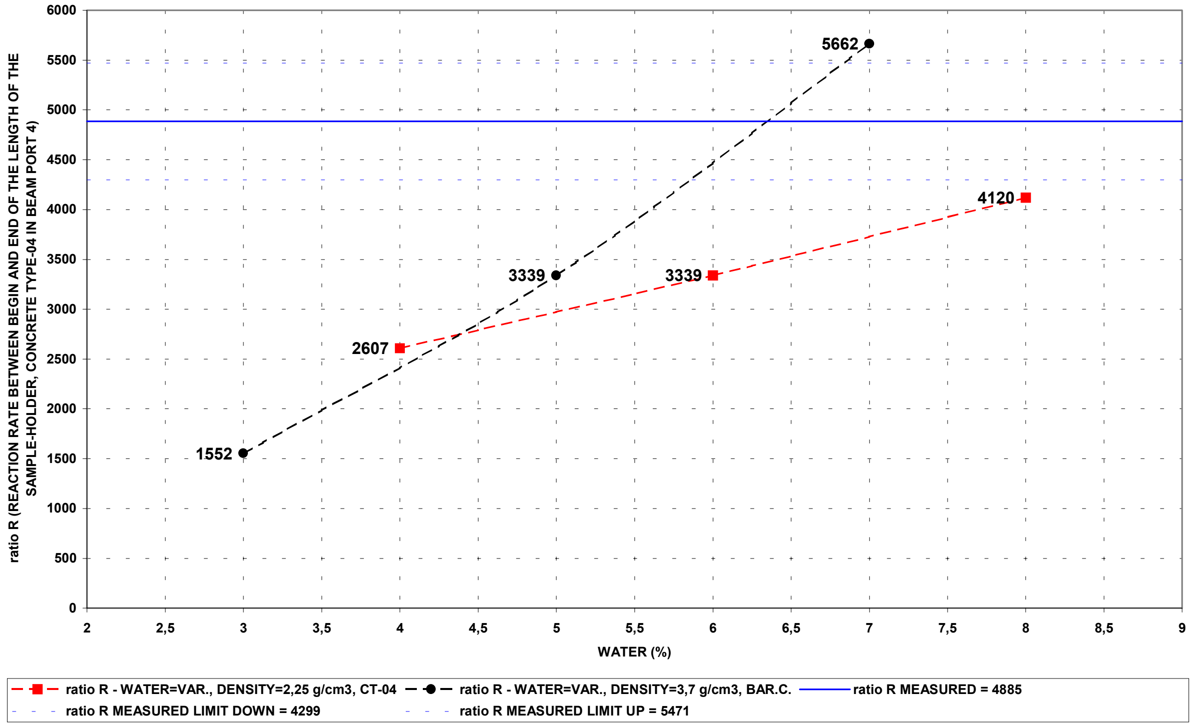

Figure 9: Altering of water content (concrete type-04, barytes concrete)

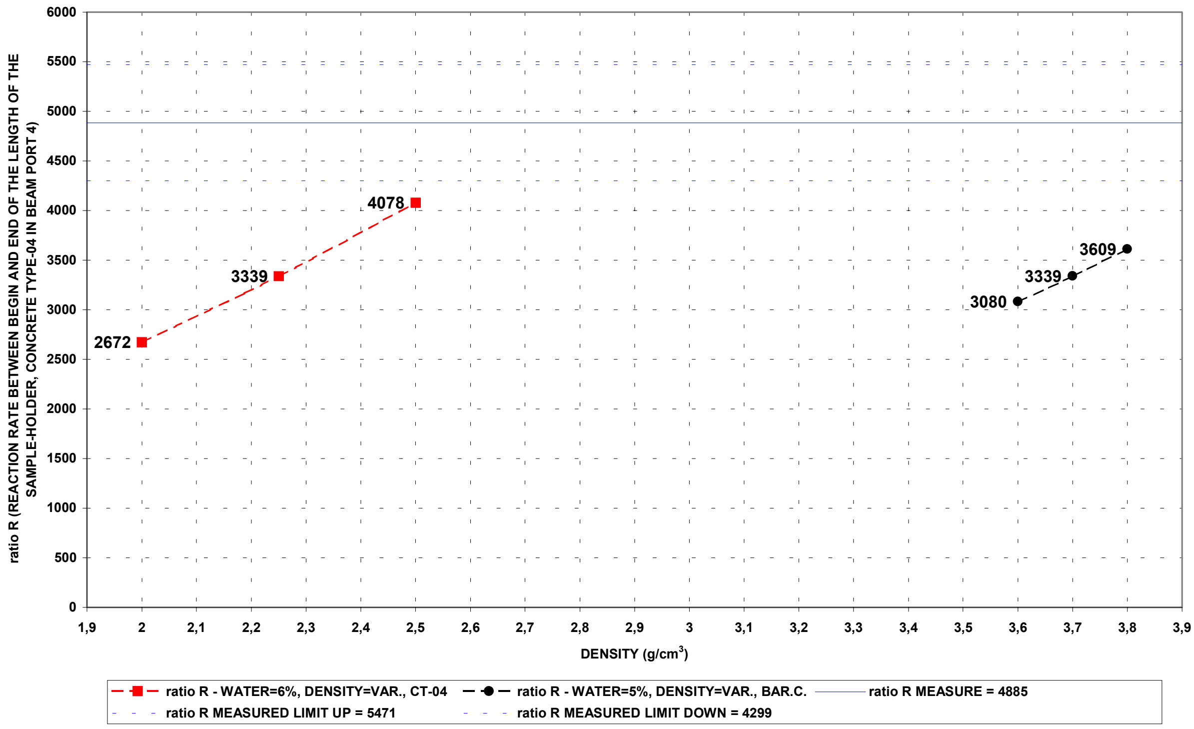

Figure 10: Altering of density (concrete type-04, barytes concrete)

Water content in barytes concrete (around of beam port 4 and also concrete type-04 in beam port 4) has a strong influence on ratio R between reaction rates over the horizontal edges of concrete type 4 longitude. It can be concluded that there is a strong impact of scattering neutrons into beam port 4 from surrounding barytes concrete. Measured ratio R is reached with calculated results at 6.35 % of water content in barytes concrete from Figure 9. Water content in concrete type 4 doesn't have strong influence on ratio R from Figure 9.

Concrete density doesn't have strong influence on ratio R. Curves for both types of concrete are almost parallel on Figure 10.
Results for calculated ratio R are in Table 4. Measured ratio R is 4885 ± 586.

Table 4: Calculated ratios R for density and water content altering with mesh 15

CONCRETE TYPE-04		**BARYTES CONCRETE**	
density = 2,25 g/cm^3	**calculated ratio R**	**density = 3,7 g/cm^3**	**calculated ratio R**
water – 4 %	2607	water – 3 %	1552
water – 6 % (reference)	3339	water – 5 % (reference)	3339
water – 8 %	4120	water – 7 %	5662
CONCRETE TYPE-04		**BARYTES CONCRETE**	
water = 6 %	**calculated ratio R**	**water = 5 %**	**calculated ratio R**
density - 2 g/cm^3	2672	density - 3,6 g/cm^3	3080
density - 2,25/cm^3 (reference)	3339	density - 3,7 g/cm^3 (reference)	3339
density - 2,5 g/cm^3	4078	density - 3,8 g/cm^3	3609

5 CONCLUSIONS

Results of calculated neutron fluxes for whole reactor body of the TRIGA Mark II research reactor are presented. TORT computer code with library BUGLE-96 (47 energy neutron groups) is used. Homogeneous model converged with mesh size - 63936 cells, so there is no need for increasing mesh size in this geometry case. Calculated parameters (neutron fluxes received with TORT code for whole reactor body with inserted beam port 4) are verified with experimental data. Agreement is very good. Final results are presented on Figures 9, 10. Calculated ratio R (ratio between reaction rates at the edges of the sample-holder, concrete type 4) is identical with measure ratio R at 6.35 % of water content in barrettes concrete. Non-homogeneous geometry is also used for determination of neutron fluxes in whole reactor body with emphasis on beam port 4 filled with polyethylene before sample-holder, concrete type-4. Homogeneous model (mesh 13) was also the basis for determination of iron activation at various distances from core center in barytes concrete in the core height size vertical length.

Iron structural mesh in biological shield of TRIGA Mark II research reactor is also considered with TORT code.
The same verified method can be applied for the equivalent scope of work in Nuclear Power Plant Krško - NEK.

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