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Comparison of Neutron Fluxes Obtained by 2-D and 3-D Geometry with Different Shielding Libraries in Biological Shield of the TRIGA MARK II Reactor

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ABSTRACT

Neutron fluxes in different spatial locations in biological shield are obtained with TORT code (TORT-Three Dimensional Oak Ridge Discrete Ordinates Neutron/Photon Transport Code). Libraries used with TORT code were BUGLE-96 library (coupled library with 47 neutron groups and 20 gamma groups) and VITAMIN-B6 library (coupled library with 199 neutron groups and 42 gamma groups). BUGLE-96 library is derived from VITAMIN-B6 library. 2-D and 3-D models for homogeneous type of problem (without inserted beam port 4) and problem with asymmetry (non-homogeneous problem; inserted beam port 4, filled with different materials) were of interest for neutron flux calculation. The main purpose is to verify the possibility for using 2-D approximation model instead of large 3-D model in some calculations. Another purpose of this paper was to compare neutron spectral constants obtained from neutron fluxes (3-D model) determined with smaller BUGLE-96 library with new constants obtained from fluxes calculated with bigger VITAMIN-B6 library. These neutron spectral constants are used in isotopic calculation with SCALE code package (ORIGEN-S). In past only neutron spectral constants determined by neutron fluxes from BUGLE-96 library were used. Experimental results used for isotopic composition comparison are available from irradiation experiment with selected type of concrete and other materials in beam port 4 (irradiation channel 4) in TRIGA Mark II reactor. These experimental results were used as a benchmark in this paper.

1 INTRODUCTION

Determination of the residual activity for reactor structure materials is one of the main tasks to solve after the reactor lifetime. 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. Neutron flux can be obtained either with experimental determination or using calculation methods. But experimental methods are practically impossible or very expensive for different internal regions of reactor body. So use of experimental activation data is limited to verification of calculated results. Purpose of using two different libraries VITAMIN-B6 [11] and BUGLE-96 [3] is to verify and compare:

- Total neutron fluxes with neutron spectra in reactor body for homogeneous type (without inserted beam port 4) in 2-D and 3-D geometry with both libraries
- Total neutron fluxes with neutron spectra in reactor body for non-homogeneous type (with inserted beam port 4 filled with different materials) in 2-D and 3-D geometry with both libraries
- Comparison of the spectral constants from two different libraries used in activity calculation using SCALE program package [7], [12] (used results from homogeneous type of the geometry for activity in biological shield)
- Consider possibility to use less computer time consuming 2-D geometry models instead of large (number of nodes) 3-D geometry models.

Calculation of neutron fluxes in biological shield (concrete) of the TRIGA Mark II research reactor in Ljubljana is performed with computer code TORT [2].

2 GEOMETRY AND CALCULATION MODEL DESCRIPTION

Geometry description can be divided in two parts:

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

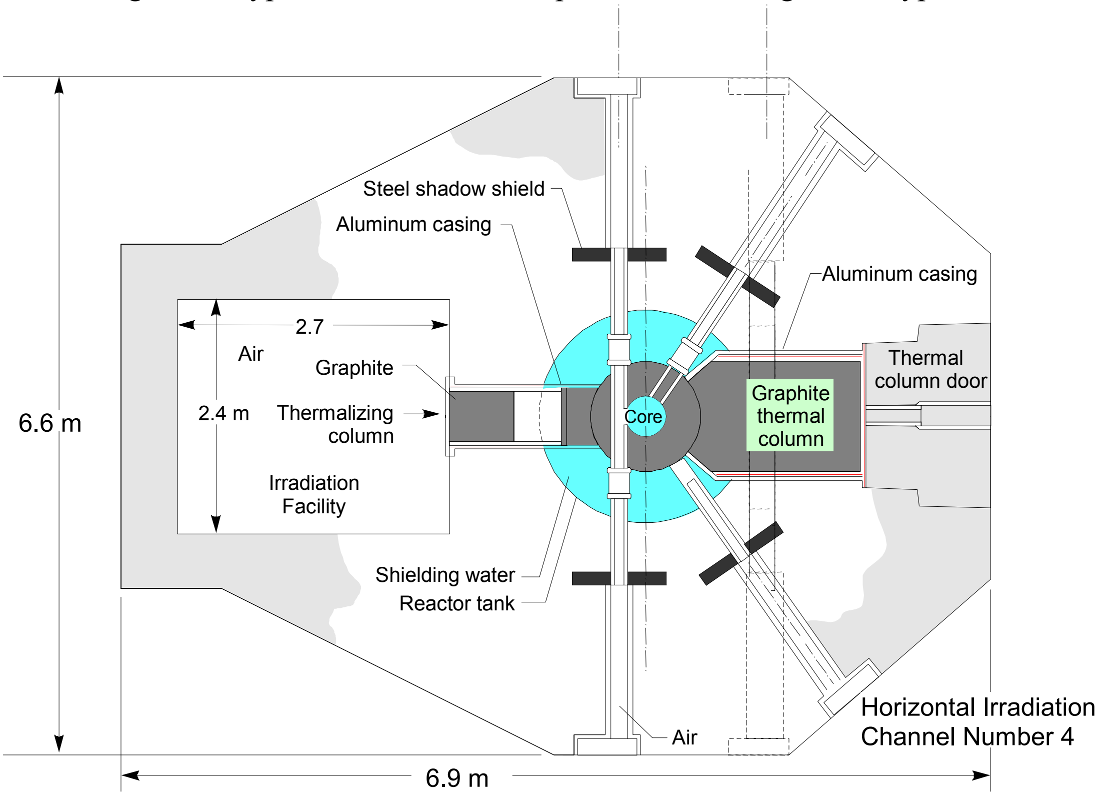

Figure 1: Horizontal cross section through TRIGA Mark II research reactor at core mid-plane

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, in r and z direction the same mesh structure was preserved) 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].
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 constructing calculation mesh 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. Homogeneous geometry type represents the basis for analysis of the spatial and energy neutron flux distribution inside whole reactor body.

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 vertical direction within the space occupied by the beam port 4 (r-direction and z-direction mesh divisions remain unchanged). Top view of the model with inserted beam port 4 is presented in Figure 2. The homogeneous model is identical to presented model on Figure 2without inserted channel 4. Whole length of beam type 4 is filled with different materials. Detailed description of materials in channel 4 is in Figure 3.

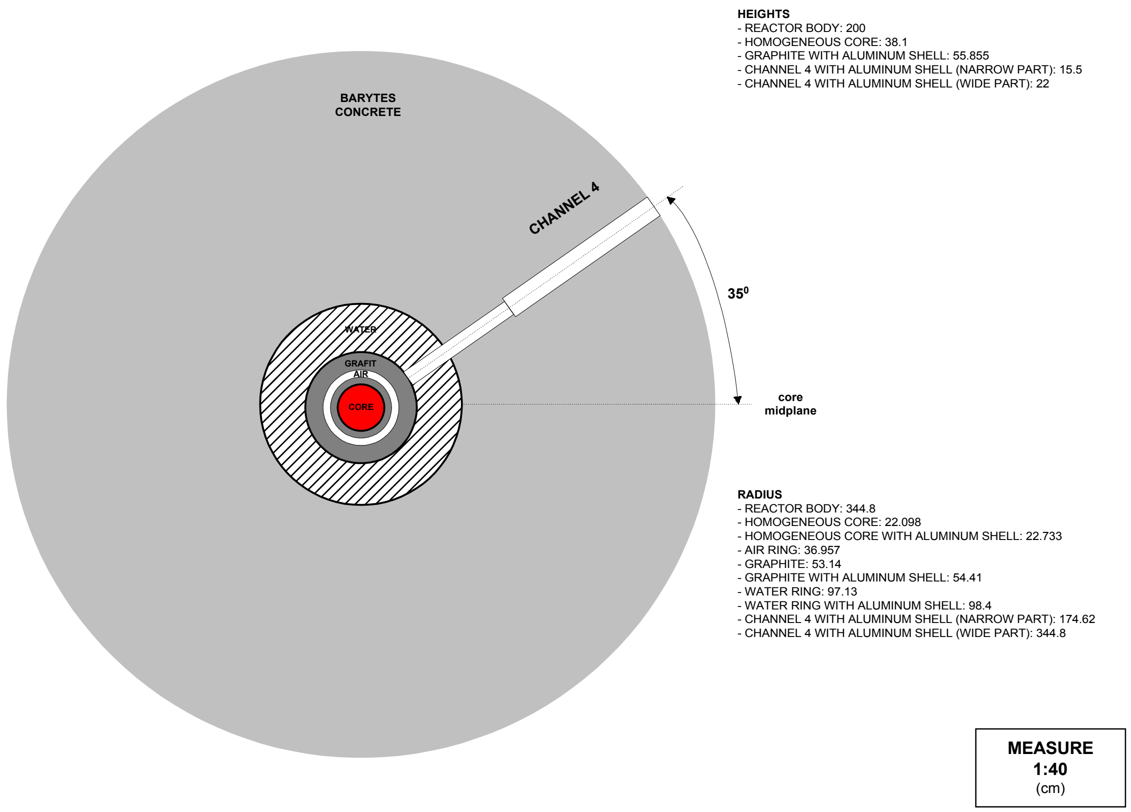

Figure 2: Schematic top view of the TRIGA Mark II research reactor model with beam port 4

Mesh size was increased from 63936 cells (homogeneous type) to 177600 cells (non-homogeneous type) for beam port 4 filled with various materials over the length.
With inserted beam port 4 in the homogeneous reactor body it is possible to:

- Make a comparison between calculated and 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

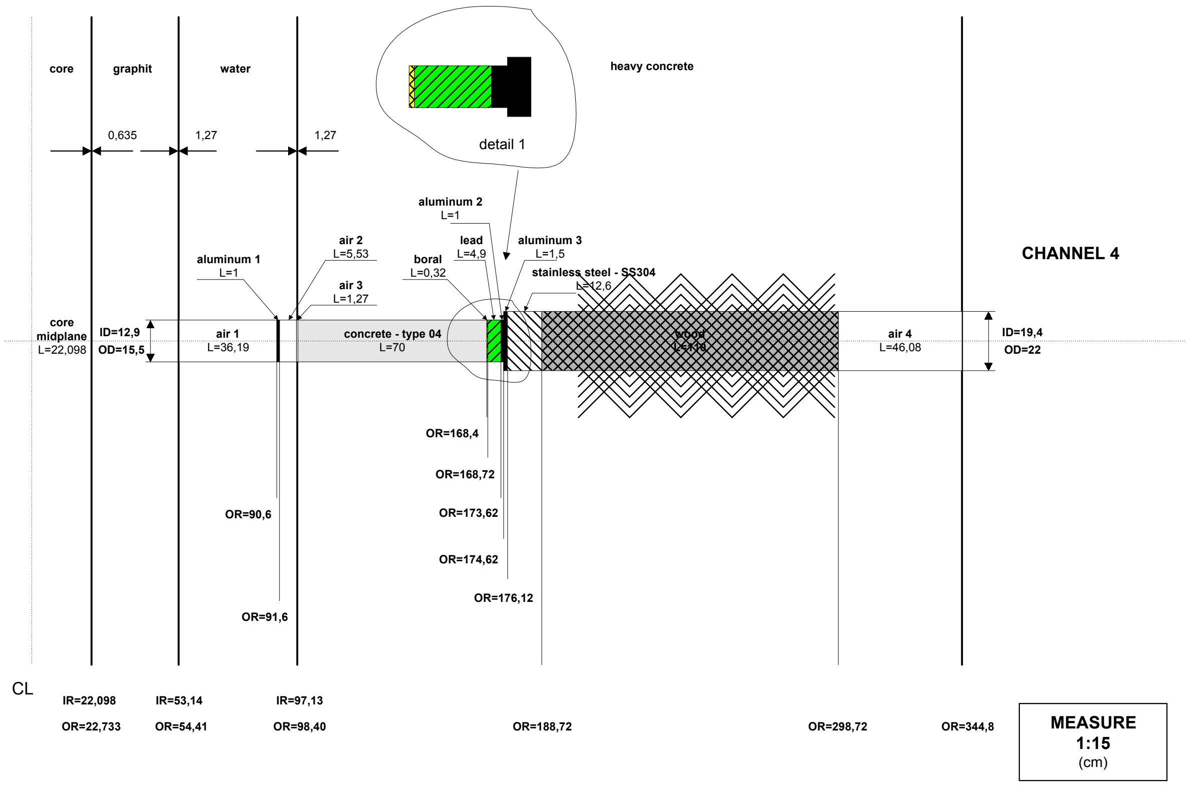

Figure 3: Beam port 4 filled with different materials during experiment (also in calculation)

3 DEVELOPMENT AND VERIFICATION OF THE CALCULATIONAL MODELS

Main goal of all consideration and comparisons is focused in testing of the homogeneous model. Calculated neutron fluxes (biological shield) from this model are used in activity calculation.
Geometrical models for 2-D and 3-D are created for homogeneous type and non-homogeneous type (with inserted beam port 4 – comparison of the calculated results with experimental results). 2-D and 3-D calculations are performed with code TORT [2].
Also libraries with macroscopic cross section are created for both libraries BUGLE-96 [3] (coupled 47 neutron groups and 20 gamma groups) and VITAMIN-B6 [11] (coupled 199 neutron groups and 42 gamma groups).

3.1 Description of the libraries used in calculations

Microscopic cross sections have to be in ANISN format [13] for code GIP [13]. Code GIP combines microscopic cross sections with material data to obtain macroscopic cross section used furthermore by code TORT [2]. It has to be executed prior using discrete transport code TORT [2] for obtaining results (neutron fluxes in whole reactor body).
Library BUGLE-96 is originally distributed in ANISN format.
Original format of VITAMIN-B6 library is not ANISN. A lot of work has to be done for converting microscopic cross section to ANISN format using program package SCALE 4.4a [12] for that conversion.
Both libraries support presentation of anisotropic scattering by Legendre expansions of arbitrary order (P_N). In this work P_3 expansion order was selected for both libraries, which have different energy group structure. The group structure for used libraries is presented in Table 1.

Table 1: Energy structure for libraries BUGLE-96 and VITAMIN-B6

	NEUTRON GROUPS		
	Thermal Range	Epithermal Range	Fast Range
	E<0.5 eV	0.5 eV < E < 1 MeV	< 1 MeV
BUGLE-96 (47 neutron groups)	47 and 46	45 to 19	18 to 1
VITAMIN-B6 (199 neutron groups)	199 to 180	179 to 63	62 to 1

3.2 Geometry models and other parameters used in calculations

Computer code TORT [2] is used for generating neutron fluxes in whole reactor body in both approximations, 2-D and 3-D. The transport process described the Boltzmann transport equation is solved using the discrete ordinates method. The directional variable is treated in S_{10} order (140 discrete directions) in all cases studied here.
Multigroup formulation treats the energy dependence. Anisotropic scattering is treated using a Legendre expansion. Reactor eigenvalue problem can also be solved. Either cylindrical (RΘZ) or cartesian (XYZ) geometry is supported.
Boundary conditions need to be proper. In our TORT input model (RΘZ geometry) we used:

- reflected boundary condition for R=0 and free (vacuum) boundary condition at outer radial limit of our model
- in Z-direction (both sides) we set free (vacuum) boundary conditions
- periodic boundary condition for Θ-direction (both sides).
    - for 90^0 symmetry reflective boundary conditions at zero and 90^0 in Θ-direction

P_3 option (scattering expansion) is used. Models are not tested regarding to various P_n choice.
Homogeneous model is denoted as reference mesh 13 (63936 cells).
Non-homogeneous model (homogeneous model with inserted beam port 4) is mesh 15 (177600 cells).

Both geometry models are described in Table 2.

Table 2: Different mesh sizes for both geometries (mesh 13, mesh 15)

<table>
<tr><th>num.:</th><th>mesh number</th><th>number of cells</th><th>R-axis
(divisions)</th><th>Θ-axis
(divisions)</th><th>Z-axis
(divisions)</th></tr>
<tr><td>1</td><td>MESH 13
reference model
(homogeneous case)</td><td>63936</td><td>74</td><td>24</td><td>36</td></tr>
<tr><td rowspan="2">2</td><td rowspan="2">MESH 15
(non-homogeneous case)
Θ-direction
(double division)</td><td rowspan="2">177600</td><td rowspan="2">74</td><td>24
(26 axial layers)</td><td rowspan="2">36</td></tr>
<tr><td>152
(10 axial layers)</td></tr>
</table>

The problem is symmetrical, so 90^0 symmetry is used for homogeneous model. In this way mesh size is decreased in great amount.
Fixed source energy spectrum is taken from procedure for creating library BUGLE-96 [3]. Source is uniformly distributed in homogeneous core volume.

4 RESULTS

4.1 Review of the selected types for calculations

Calculations are executed in the configurations presented in Table 3:

Table 3: Types of calculations

num.:	Library	Type of geometry (dimensional)	Type of model
1	BUGLE-96	2-D	Homogeneous
2	BUGLE-96	2-D	Non-homogeneous
3	BUGLE-96	3-D	Homogeneous
4	BUGLE-96	3-D	Non-homogeneous
5	VITAMIN-B6	2-D	Homogeneous
6	VITAMIN-B6	2-D	Non-homogeneous
7	VITAMIN-B6	3-D	Homogeneous
8	VITAMIN-B6	3-D	Non-homogeneous

All input models with both libraries have to be created with proper data. Libraries must have correct input data with mixtures used in problem and selected microscopic cross sections.

Non-homogeneous model is filled with air in first part of beam port 4 (see Figure 3) for all types of calculations included in Table 3.
Calculated neutron fluxes normalized to measured value are presented in Figure 4. The fluxes were normalised to experimental total flux value measured in core location F19 (near outer limit of core region, at 21 cm). Results are obtained with BUGLE-96 library for homogeneous model, non-homogeneous model filled with air in first part of the beam port 4 (see Figure 3) and non-homogeneous model filled with polyetilen instead of the air. Non-homogeneous results are used for verifications with experimental data. Activity of the biological shield is calculated using results from the homogeneous case.

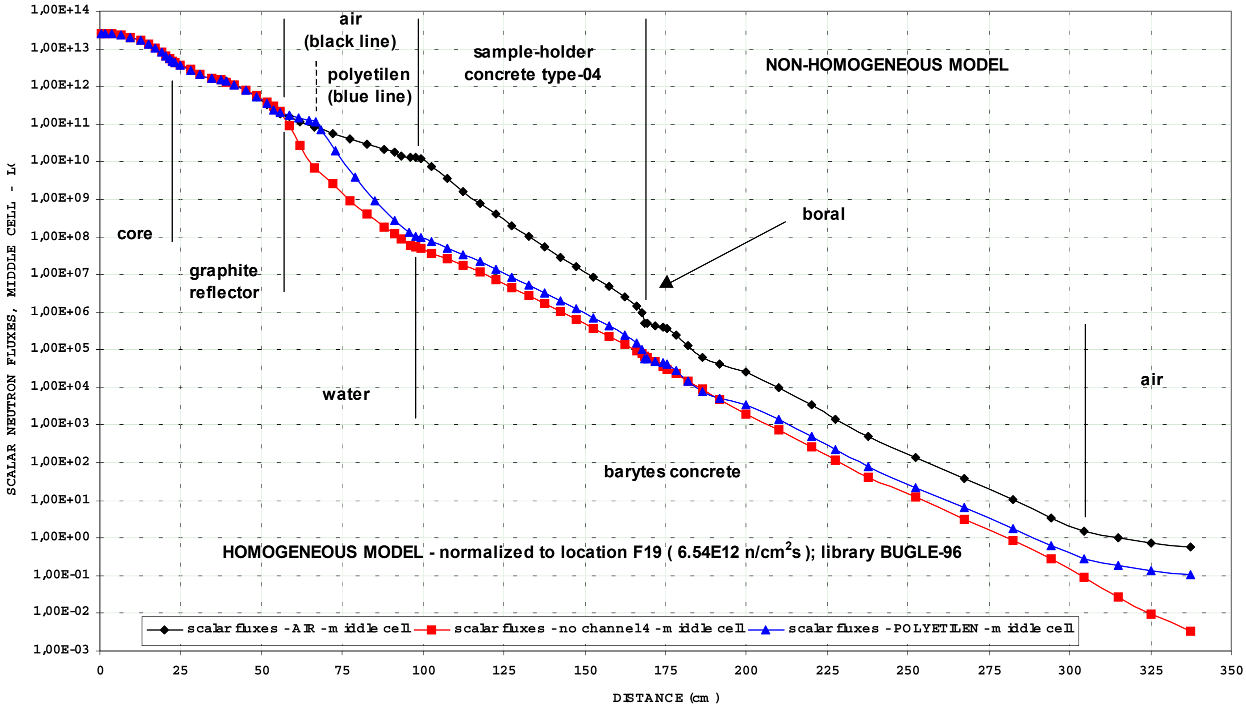

Figure 4: Normalized calculated neutron fluxes used in verification of the model and activity calculation

4.2 Comparisons between calculations specified in Table 3

General observation between 2-D and 3-D results (using the same library) is that scalar neutron fluxes (normalized, see Figure 5, 6) obtained with 2-D models are larger than results with 3-D models. 2-D model has only one axial plane with different boundary conditions as 3-D model. Results are presented for axial plane equal to core-mid plane for 2-D model. Anisotropic scattering from axial planes and surrounding cells into the observing cell is present in the 3-D model. 2-D model has reflected boundary condition at top and bottom direction, so the neutron flux is hold within one axial plane. There is no anisotropic scattering from other axial planes in 2-D geometry, so total neutron fluxes are greater in 2-D models.

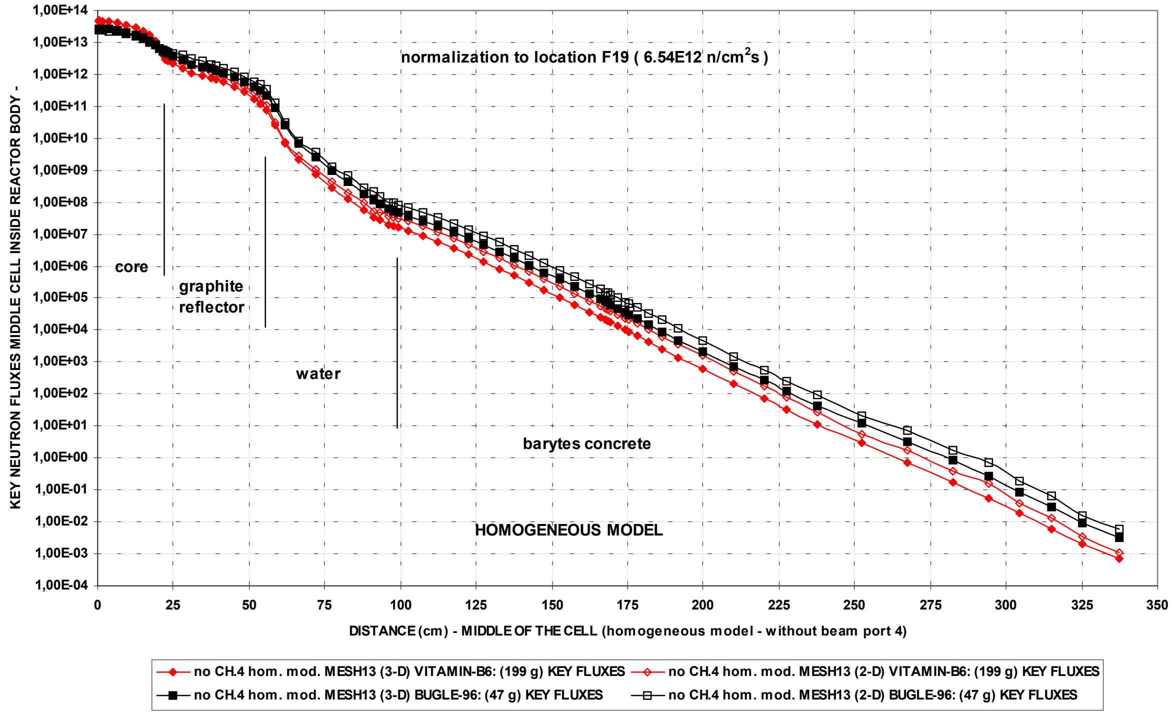

Figure 5: Various types of calculations for the homogeneous geometry (see Table 3)

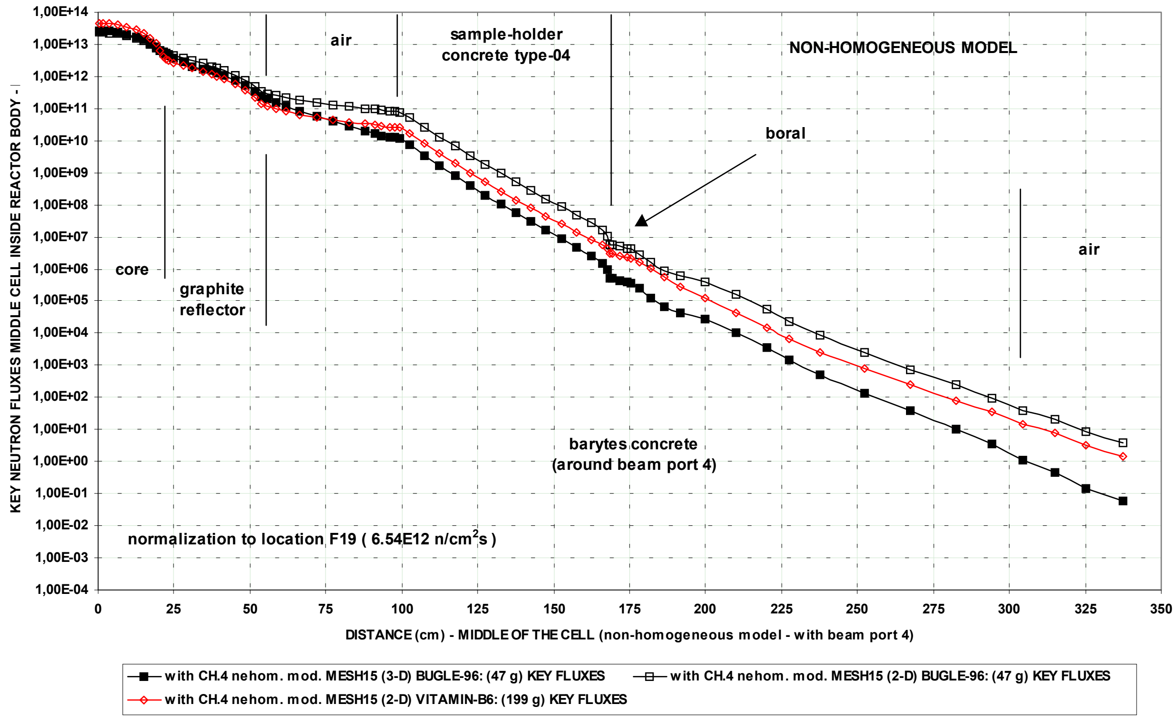

Figure 6: Various types of calculations for the non-homogeneous geometry (see Table 3)

Gap between total neutron fluxes (normalized, see Figure 5, 6) comparing 2-D and 3-D (the same library) results increases approximately to the barytes concrete for both geometrical models. Normalized neutron flux from library VITAMIN-B6 decreases (after core region) regarding to library BUGLE-96 in the high moderator regions (graphite and water ring) due to more detailed representation of the thermal groups. Upscattering is also included (36 thermal groups in VITAMIN-B6). This feature is more sensible in the non-homogeneous geometry (Figure 6).
Non-homogeneous model with VITAMIN-B6 (item 8 in the Table 3) is not included in the Figure 6 due to the problems, which arise during the execution with TORT code. Problems are connected with internal arrays in the TORT code because of the huge amount of the specified input data.
From comparison of 2-D or 3-D results one can see, that the best representation for the geometry of interest (in our case is homogeneous type – subsequent calculation of activity in the biological shield) are results obtain with 3-D geometry, using library VITAMIN-B6.
After performing normalization (see Figure 4), total neutron fluxes can be used for the activity calculations and spectral constants determination [7] (THERM ,RES, FAST).
Correct normalized total neutron fluxes with defined spectral constants [7] for homogeneous model are of the main interest. These results are used for activity calculation of the biological shield, which is the main purpose of this work

4.3 Neutron spectra for homogeneous model (biological shield) using both libraries

Neutron spectra in homogeneous model is used as input data by using various spectral constants THERMAL, RES, FAST [7], [12]. Beside good known mixtures used in the model another important object is neutron flux.

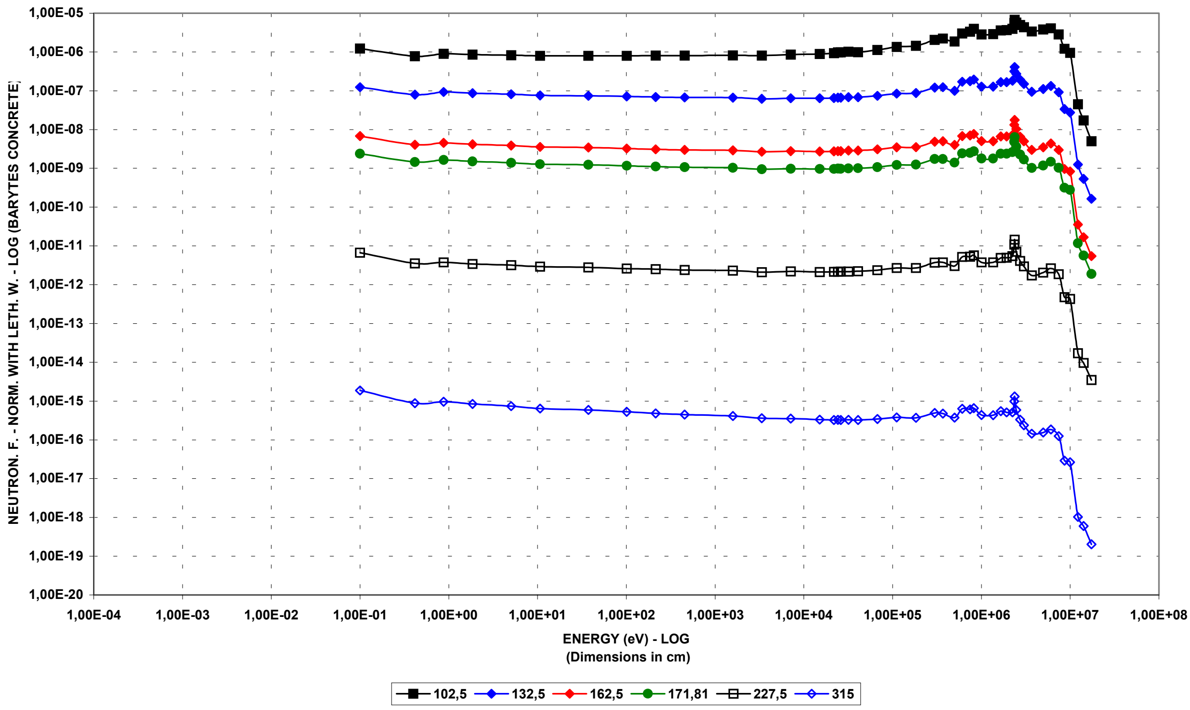

Figure 7: Neutron spectra in biological shield at core mid-plane using BUGLE-96 (3-D)

One can use both libraries BUGLE-96 and VITAMIN-B6. Neutron spectra normalized with lethargy width is bigger with BUGLE-96 than with using library VITAMIN-B6. This is also visible from Figure 5, which represents total neutron flux across the radius in the core-mid plane height.

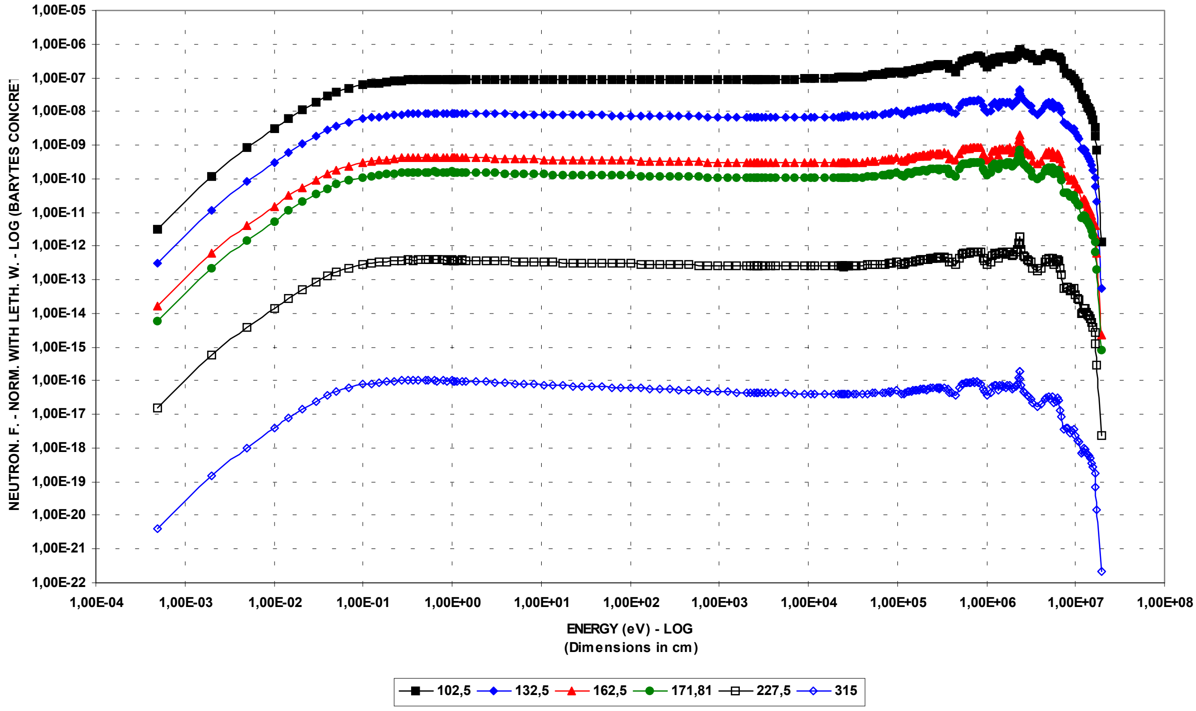

Figure 8: Neutron spectra in biological shield at core mid-plane using VITAMIN-B6 (3-D)

BUGLE-96 is derived from VITAMIN-B6. Thermal spectra of BUGLE-96 comparing with VITAMIN-B6 is different (see Table 1). BUGLE-96 has only two energy groups in thermal range (below 0.5 eV). VITAMIN-B6 has 20 thermal energy groups (below 0.5 eV). VITAMIN-B6 better represents thermal range. BUGLE-96 represents neutron spectra in fast and epithermal range almost as VITAMIN-B6 from Figure 8 (above 1 eV).
Spectral constants used as input data for activity calculation [7], [12] for biological shield are presented for both libraries in table below. Spectral constants are used with code ORIGEN-S [7] for different radius at the core mid-plane.

Table 4: Spectral constants calculated for different radius at the core mid-plane

	VITAMIN-B6				BUGLE-96			
Radius	**102.5cm**	**162.5 cm**	**227.5 cm**	**315 cm**	**102.5cm**	**162.5 cm**	**227.5 cm**	**315 cm**
FAST	6.383E+0	1.866E+0	1.523E+0	5.660E-1	9.251E-1	2.302E-1	1.600E-1	5.327E-2
RES	5.875E-1	3.871E-1	3.576E-1	2.497E-1	8.740E-2	5.196E-2	4.198E-2	2.766E-2
THERMAL	4.727E-1	4.732E-1	4.754E-1	4.796E-1	4.800E-1	4.814E-1	4.837E-1	4.857E-1

Trend is generally the same for each spectral constant comparing both libraries. THERMAL is almost the same for both libraries. FAST and RES have the same decreasing trend across the radius. Rate between FAST or RES is approximately one decade comparing both libraries across radius. Differences for spectral constants arises due to different energy group structure regarding both libraries and also due to algoritem for those spectral constants. Thermal flux is greater obtained with BUGLE-96 regarding to VITAMIN-B6 (below 0.5 eV), it is used in denominator for all spectral constants, which is the reason for lower values of the FAST and RES for BUGLE-96 [7], [12]. This can be concluded observing Figure 7 and Figure 8 (thermal range).
Calculations are made for the same input thermal neutron flux for cases enclosed in Table 4. Irradiation is 1 year. Spectral constants THERM, RES, FAST are changed for VITAMIN-B6 library (new set of results) comparing to BUGLE-96 library (old set of results – original results). Results are obtained with ORIGEN-S [7].

Table 5: Comparison for influence of spectral factors (Table 4) with the same thermal flux

IRRADIATION 1Y (end at time=0), RESULTS FOR RADIOACTIVITY [Ci] - DECAY after IRRADIATION

RESULTS - BIOLOGICAL SHIELD - HOMOGENEOUS MODEL-3D (VARIOUS RADIUS) AT THE CORE MID-PLANE

VITAMIN-B6

		time=0	1. yr	10. yr	30. yr	100. yr	300. yr
R=102.5cm	totals	8.51E+00	5.03E-03	5.99E-04	1.13E-04	3.79E-06	2.06E-06
R=162.5cm	totals	1.08E-02	1.41E-05	1.99E-06	3.93E-07	9.89E-09	5.02E-09
R=227.5cm	totals	8.18E-06	1.21E-08	1.74E-09	3.46E-10	8.46E-12	4.24E-12
R=315.0cm	totals	9.69E-10	2.50E-12	3.88E-13	7.94E-14	1.76E-15	8.45E-16

BUGLE-96

		time=0	1. yr	10. yr	30. yr	100. yr	300. yr
R=102.5cm	totals	1.51E+00	2.91E-03	4.82E-04	1.04E-04	2.27E-06	1.05E-06
R=162.5cm	totals	2.38E-03	9.90E-06	1.72E-06	3.73E-07	7.64E-09	3.41E-09
R=227.5cm	totals	1.81E-06	8.66E-09	1.52E-09	3.31E-10	6.71E-12	2.97E-12
R=315.0cm	totals	3.14E-10	1.98E-12	3.52E-13	7.70E-14	1.54E-15	6.76E-16

Results with higher spectral constants RES and FAST (see Table 4) for VITAMIN-B6 library are higher than values obtained with BUGLE-96 (THERM spectral constant is approximately the same for this calculated case). From the comparisons of normalized total neutron fluxes (to the location F19 in the core) for both libraries one can conclude, that total activities are 5 to 6 times higher with spectral constants derived from neutron fluxes calculated with VITAMIN-B6 library. Absolute total neutron fluxes (see figure 5) for homogeneous model are approximately 7 times greater with BUGLE-96 library. The net effect of contribution to the calculated activity of biological shield are roughly zero. The activity of the biological shield presented in [14] remains unaffected using bigger library VITAMIN-B6.

5 CONCLUSIONS

Results of calculated neutron fluxes for whole reactor body of the TRIGA Mark II research reactor are presented for homogeneous and non-homogeneous model with 2-D and 3-D ceases and two libraries VITAMIN-B6 and BUGLE.-96. Main goal is to verify homogeneous model with 3-D case, which is used for qualification of the activity in the biological shield. Results are practically the same for both libraries due to opposite effects of the spectral constant (particularly RES, FAST for VITAMIN-B6, higher, comparing to BUGLE-96) and normalized total neutron fluxes for all reactor body (higher with BUGLE-96, comparing to VITAMIN-B6). Activities published in [14] are the approximately the same for both libraries, considering zero net effect of the two opposite facts.

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