Fuel Element Burnup Determination in HEU - LEU Mixed TRIGA Research Reactor Core

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Abstract

This paper presents the results of a burnup calculations and burnup measurements for TRIGA FLIP HEU fuel elements and standard TRIGA LEU fuel elements used simultaneously in small TRIGA Mark II research reactor in Ljubljana, Slovenija. The fuel element burnup for approximately 15 years of operation was calculated with two different in house computer codes TRIGAP and TRIGLAV (both codes are available at OECD NEA Data Bank). The calculation is performed in one-dimensional radial geometry in TRIGAP and in two-dimensional (r,φ) geometry in TRIGLAV. Inter-comparison of results shows important influence of in-core water gaps, irradiation channels and mixed rings on burnup calculation accuracy. Burnup of 5 HEU and 27 LEU fuel elements was also measured with reactivity method. Measured and calculated burnup values are inter-compared for these elements

Introduction

The methods for determining fuel element burnup have recently become interesting due to the activities related to the shipment of US originating highly enriched fuel elements back to US for final disposal before 2006. Many research reactor operators are faced with the problem of determining the burnup of the fuel elements that have been used in mixed and non-uniform core configurations with long and complicated operational history. The most common and practical method for determining fuel element burnup in research reactors is reactor calculation [1]. Other methods, e.g. gamma scanning [2], are normally too complicated for determining burnup of hundreds of fuel elements in relatively short time as it is the case in the preparation activities for the shipment operation.

The purpose of this paper is to analyze the accuracy of the burnup calculation typically performed by the operators of small TRIGA reactors who normally use simple, widely available fuel management codes based on rough physical models. Attention is paid to the effects, which are frequently neglected in simple, often one-dimensional fuel management calculations. The effects are analyzed by inter-comparing results of one-dimensional and two-dimensional burn-up calculations and by comparing them to the measurements.

Description of the Burnup Calculation

The burnup calculations were performed using TRIGAP and TRIGLAV computer packages, which are available at the OECD Nuclear Energy Agency Data Bank, Saclay, France. Detailed description of both codes can be found in program manuals [3] and [4]. Only brief description of their functions relevant for understanding of the physical models used in burnup calculations is presented here.

Both codes are developed for TRIGA reactor geometry with annular fuel element rings. The calculation is performed in one-dimensional radial geometry in TRIGAP and in two-dimensional r-φ geometry in TRIGLAV. Multi-group diffusion approximation is used in both codes: two groups in TRIGAP and four groups in TRIGLAV. The multi-group effective cross-sections are in both codes calculated in unit-cell approximation using WIMS-D/4 [5] lattice code. The original WIMS cross-section library extended for TRIGA reactor specific nuclides (hydrogen bound in zirconium lattice, erbium) is used.

Approximately the same unit-cell model is used in TRIGAP and TRIGLAV. Each ring is subdivided into unit cells as presented in Figure 1. Normally one unit cell represents a fuel rod in the center surrounded by water. The fuel rods may be different in uranium content and enrichment. Three types of TRIGA fuel elements are treated: two standard fuel elements (LEU type) containing 20% enriched uranium and FLIP fuel element containing 70% enriched uranium (all TRIGA fuel elements contain only one fuel rod). More detailed specification of the fuel elements treated in the paper may be found in Table 1.

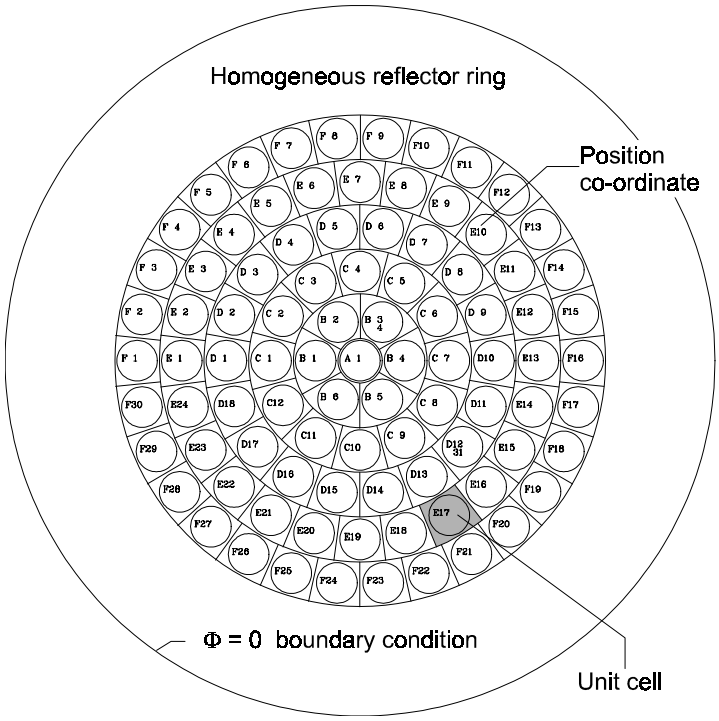

Figure 1. TRIGLAV geometry model of TRIGA reactor core. Homogeneous unit cells are assumed. Homogeneous reflector ring radius is not drawn to scale.

The homogenized unit-cell cross sections are calculated for each fuel element separately as a function of burnup, fuel temperature, water temperature and xenon concentration. Instead of the fuel rod, the unit cell may contain the control rod, the irradiation tube, the graphite rod, the neutron source or only water depending on the core configuration. In this case the cross sections are calculated by surrounding the unit-cell with fuel unit-cells and by homogenizing only the inner part of the system belonging to the particular non-fuel unit-cell.

Main difference between TRIGAP and TRIGLAV appears after the unit cell calculation. The group constants belonging to different unit cells are homogenized in TRIGAP by volume weighting over each ring and the diffusion calculation is performed in one-dimensional radial

geometry. In TRIGLAV the rings are not smeared and the diffusion calculation is performed in two-dimensional geometry.

Table 1. TRIGA fuel elements considered in calculations.

	Standard 8.5%	Standard 12%	FLIP
dimensions			
fuel element length [cm]	72.06	72.06	72.06
fuel meat length [cm]	38.10	38.10	38.10
fuel element diameter [cm]	3.75	3.75	3.75
fuel meat diameter [cm]	3.64	3.64	3.64
composition[(a)]			
fuel meat	U-ZrH	U-ZrH	U-ZrH-Er
cladding	SS	SS	SS
mass of U-ZrH [g]	2235	2308	2259
uranium concentration [%]	8.5	12	8.5
m(U) [g]	190	277	192
enrichment [%]	20	20	70
m(^{235}U) [g]	38	55	134
burnable poison	-	-	Er
erbium concentration [%]	-	-	1.5

[(a)]Typical fuel element composition data. Particular fuel element composition data can differ from values in this table.

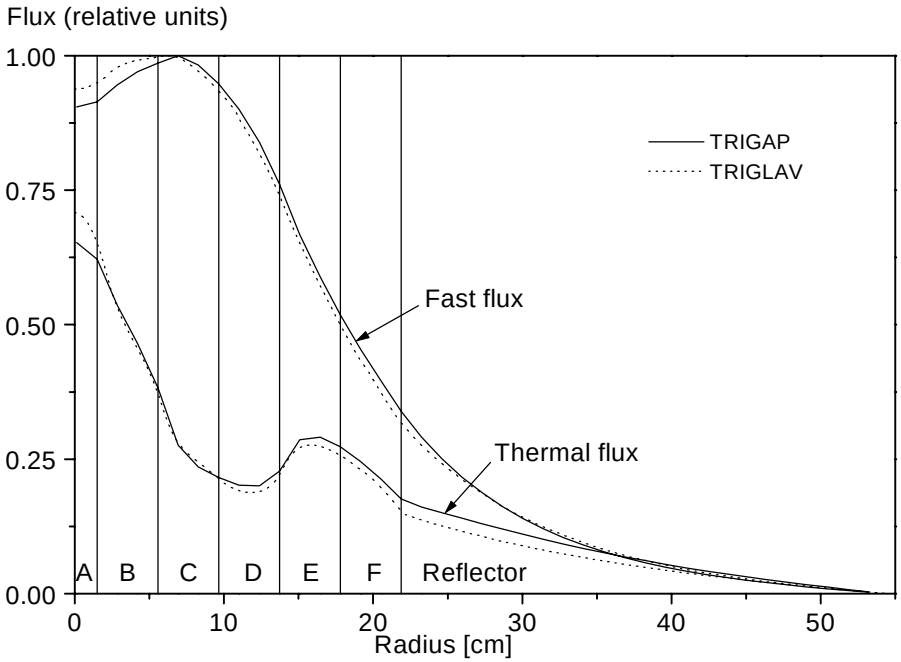

Figure 2. Relative thermal and fast flux radial distributions for operating core 130 calculated with TRIGAP and integrated two-group radial distributions calculated with TRIGLAV (group boundary at 0.625 eV).

TRIGAP package is based on two-group diffusion equation (group boundary at 1 eV) in one-dimensional cylindrical geometry. It is solved in the finite differences approximation by fission density iteration method. The physical model of TRIGAP is not appropriate for the problems involving strong spectral and spatial variations of neutron flux distribution due to the two-group approximation and ring homogenization. It is appropriate for simple compact uniform loading patterns with only one type of fuel elements in the same ring. It fails to predict correct burnup for mixed rings or for regions near control rods and in-core irradiation channels. Typical two-group flux distribution calculated with TRIGAP is presented in Figure 2. The radial flux distribution calculated with TRIGLAV and integrated in the angular direction is added for comparison. The core configuration is realistic. It corresponds to core 130 of the Jožef Stefan Institute reactor and is schematically presented in Figure 3

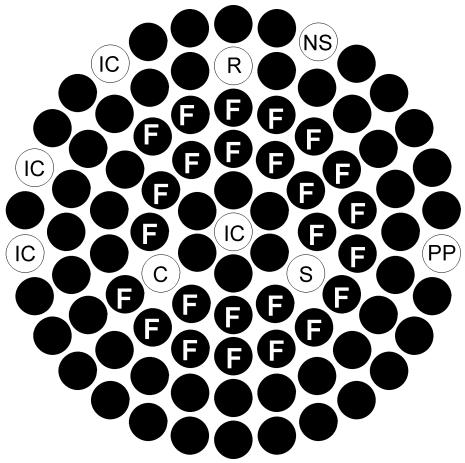

Figure 3. Schematic drawing of core loading pattern 130. The black circles denote the standard fuel elements. The white F in the black circle indicates FLIP fuel element. S, C and R denote the positions of the control rods (safety, shim and regulating, respectively). IC denotes the location of the irradiation channel; PP is the pneumatic post irradiation position; and NS is the neutron source position.

TRIGLAV code is based on four-group diffusion equation in two-dimensional r-φ cylindrical geometry. Method of solution is finite differences and fission density iteration. The geometry model represents full TRIGA cylindrical core with maximum 7 rings and reflector (see Figure 1). The finite differences grid is part of the model. It is not equidistant, as it has to adapt to the unit cell boundaries that do not coincide at the ring boundaries. Every location either occupied by the fuel element, the control rod, the irradiation channel or left empty is treated explicitly as a homogeneous region equivalent to a unit cell. The unit cell group constants are calculated with program WIMS-D/4, which is integrated in the original package. Power density distribution calculated with TRIGLAV is presented in Figure 4 for the same core loading pattern as presented in Figure 3 (core no. 130).

By comparing results of one-dimensional and two-dimensional calculations presented in Figure 2 and Figure 4, respectively, we can get a qualitative estimate of the effects influencing burnup distribution in realistic mixed cores. Large calculation errors may be expected particularly in the regions of large flux gradients e.g. in the mixed rings, in the vicinity of water gaps, irradiation channels and close to the control rod. Relative power density distribution calculated with TRIGLAV and presented in Figure 4 shows considerably lower power density in the part of mixed ring D occupied with standard fuel elements. Opposite effect may be seen in the vicinity

of the control rod followers' positions. There good moderation in the water increases thermal flux level in the surrounding. It may be noted that the control rods are considered completely withdrawn in this calculation which means that their positions in the core are filled with water.

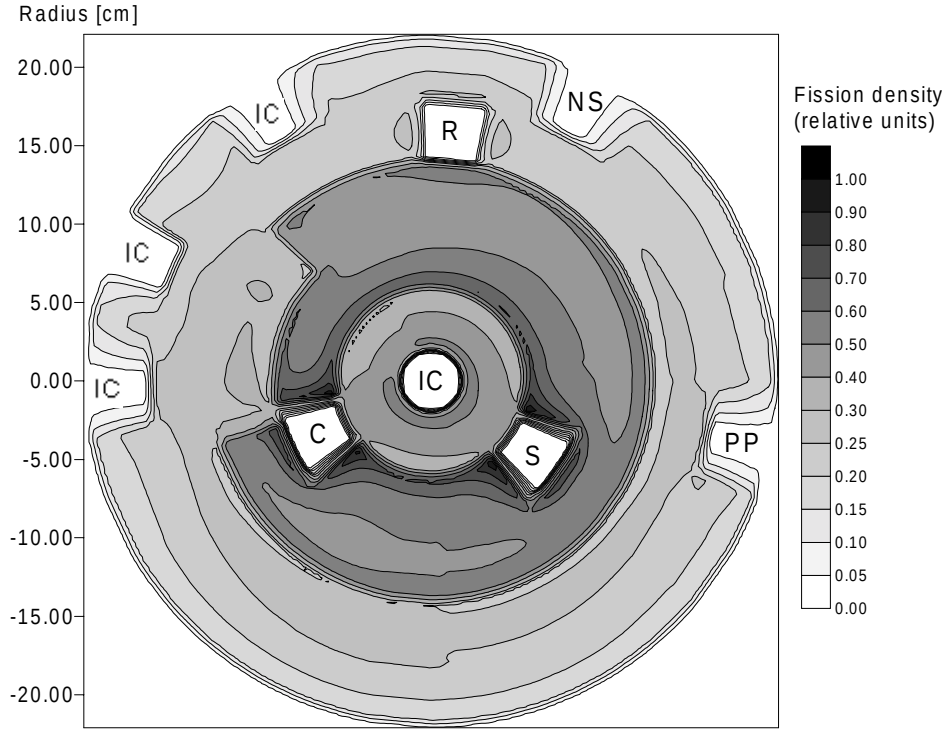

Figure 4. Relative power density distribution calculated with TRIGLAV for operating core 130. Schematic drawing of core loading pattern 130 is presented in Figure 3.

Brief description of reactor operating history

Fuel elements operating history was recalculated from 1983 up to this year. Standard 20% enriched uranium fuel elements with 8.5w% uranium were used in reactor core together with FLIP (8.5w% uranium, 70% enriched) fuel elements from 1983 up to 1989. These cores were mixed with LEU fuel elements in B, E and F rings, with FLIP fuel elements in C ring and with mixed D ring. Typical example for such mixed core configuration is core 130 presented in Figure 3. In year 1990 reconstruction of the reactor core was made, when new grid plates were installed together with new transient rod. After the reconstruction, only standard LEU fuel was used in reactor.

Burnup Calculation Results

Comparison of fuel element burnup calculated with TRIGAP and TRIGLAV is presented in Table 2 for three groups of elements of different types. These elements were selected for measurements out of more than 150 fuel elements that were treated in the calculations. We used the same input data for both programs. The elements that spent the main parts of their lives at the positions of the two-dimensional effects are denoted in the comments. The difference between the one- and two- dimensional results exceeds 5% for such elements.

The biggest differences are observed for standard fuel elements with 8.5% uranium (e.g. fuel elements 6163, 6170 and 7844) which spent part of their operating time in mixed D ring together

with highly enriched FLIP fuel elements. The TRIGAP calculation based on homogenized mixed rings systematically over-predicts the burnup of LEU fuel elements and under-predicts the burnup of HEU fuel elements (e.g. fuel elements 8307 and 8680) in mixed D ring. The one-dimensional model also fails to predict the correct burnup for the fuel elements that spent part of their lives near water gap, irradiation channel or control rod. These effects can be observed for standard elements 7243, 7247 and 7258, and for FLIP fuel element 8681 that spent most of its live in the vicinity of control rod.

Table 2. Comparison of burnup calculated in 1D (TRIGAP) and 2D (TRIGLAV) approximation for selected LEU and HEU fuel elements.

Element Number	m(^{235}U) [g] / Er [%]	Type	Calculated Burnup		Difference [%]	Reason of Discrepancy[(a)]
			1D [MWd]	2D [MWd]		
6094	39 / --	Standard fuel 8.5% uranium concentration, 20% enriched	9.126	9.177	0.6	
6097	38 / --		7.097	7.170	1.0	
6163	38 / --		8.697	7.294	17.6	mixed (D6)
6170	39 / --		7.755	6.761	13.7	mixed (D7)
6174	38 / --		8.742	8.840	1.1	
6175	38 / --		8.927	8.858	0.8	
6189	38 / --		8.394	8.357	0.4	
6197	39 / --		7.932	8.077	1.8	
6198	38 / --		8.594	8.731	1.6	
6200	38 / --		8.279	8.394	1.4	
7841	38 / --		7.237	7.057	2.5	
7844	38 / --		7.636	6.300	19.2	mixed (D5)
7849	38 / --		6.375	6.524	2.3	
8528	38.20 / --		7.709	7.638	0.9	
8533	37.49 / --		8.237	8.074	2.0	
8535	37.83 / --		8.073	7.746	4.1	
8536	37.45 / --		8.505	8.439	0.8	
8537	38.02 / --		8.666	8.615	0.6	
6945	55.47 / --	Standard fuel 12% uranium concentration, 20% enriched	0.546	0.549	0.6	
7213	55.31 / --		1.950	2.015	3.3	
7215	54.97 / --		0.009	0.008	-	
7219	55.47 / --		1.950	1.977	1.4	
7220	55.42 / --		1.950	1.971	1.1	
7228	55.31 / --		1.574	1.587	0.8	ring with
7243	55.00 / --		0.277	0.223	21.6	water gap (F5)
7247	55.63 / --		1.574	1.719	8.8	near CR (D5)
7258	54.70 / --		0.521	0.477	8.8	near IC (E15)
7478	134.21 / 1.48	FLIP fuel 8.5% uranium concentration, 70% enriched	12.757	13.314	4.3	
8307	135.08 / 1.52		11.218	12.368	9.8	mixed (D17)
8318	132.80 / 1.47		13.191	13.868	5.0	
8680	136.31 / 1.64		8.766	9.494	8.0	mixed (D18)
8681	134.16 / 1.64		11.033	11.896	7.5	near CR (C4)

[(a)]Reason of discrepancy (prevailing position in the core during burnup).

Comparison of Calculated and Measured Burnup

The burnup of all fuel elements was calculated with both codes TRIGAP and TRIGLAV. The burnup of selected fuel elements of all types was also measured by the reactivity method [1]. The measurement procedure is described in details in reference [6], only the most important steps of the procedure are described as follows.

Reactivity measurements were performed in homogeneous reactor core loaded only with LEU fuel. Reactivity worth of measured fuel elements in selected position was measured with a digital reactivity meter [7]. Several fuel elements with well-defined burnup were included in the measurements to obtain reference values for experimental relation between burnup and reactivity. The theoretical burnup dependence of reactivity for each measured fuel element type was also calculated with TRIGLAV code. Experimental relation between burnup and reactivity was than obtained combining both the theoretical and reference reactivity values. On the basis of measured fuel element reactivity and these experimental relations we calculated burnup presented in Table 3.

Two different sources of experimental errors were identified. The first includes the uncertainties related to the measurement of the fuel element reactivity, small uncertainties of fuel element positioning within their locations and small movements of the control rods. We estimated that these uncertainties contribute less than 50% to cumulative error. The second source of experimental uncertainty related to reactivity method is the fuel element composition variations. Our calculations [8] showed that the H/Zr ratio variation in QA acceptable range can contribute to reactivity change on the order of $\Delta k/k = 30\times10^{-5}$. This can be equivalent to reactivity loss due to 1.5-MWd burnup.

Table 3. Measured reactivity and burnup values for standard and FLIP fuel elements considered in the paper.

Element Number	Measured Reactivity ($\Delta k/k$) [10^{-5}]	Measured Burnup [MWd]	Element Number	Measured Reactivity ($\Delta k/k$) [10^{-5}]	Measured Burnup [MWd]
6094	79	9.0 ± 0.5	8536	82	8.9 ± 0.5
6097	138	7.0 ± 0.5	8537	78	9.0 ± 0.5
6163	150	6.6 ± 0.5	6945	92	1.5 ± 0.6
6170	158	6.3 ± 0.5	7213	78	2.2 ± 0.7
6174	87	8.7 ± 0.5	7215	145	0.0 ± 0.2
6175	64	9.5 ± 0.5	7219	97	1.3 ± 0.5
6189	125	7.4 ± 0.5	7220	95	1.4 ± 0.5
6197	130	7.3 ± 0.5	7228	81	2.1 ± 0.7
6198	92	8.6 ± 0.5	7243	136	0.2 ± 0.2
6200	105	8.1 ± 0.5	7247	89	1.6 ± 0.7
7841	116	7.8 ± 0.5	7258	131	0.3 ± 0.2
7844	148	6.7 ± 0.5	7478	68	14.3 ± 1.0
7849	157	6.4 ± 0.5	8307	34	12.7 ± 1.0
8528	100	8.3 ± 0.5	8318	70	14.4 ± 1.0
8533	91	8.6 ± 0.5	8680	-14	10.6 ± 1.0
8535	115	7.8 ± 0.5	8681	-48	9.0 ± 1.0

The experimental results are presented in Figure 5, Figure 6 and Figure 7 for two types of standard LEU fuel elements and for HEU FLIP fuel elements, respectively. All experimental results are compared to one-dimensional (TRIGAP) and two-dimensional (TRIGLAV) calculated burnup values. Comparison of measured and calculated results shows that two-dimensional calculations better predict the burnup for almost all fuel elements. Figure 5 and Figure 7 show that TRIGLAV better predicts burnup of mixed rings containing HEU and LEU fuel elements than one-dimensional calculations (e.g. fuel elements 6163, 6170, 7844, 8307 and 8680).

Two-dimensional calculations failed to reduce the discrepancy between measurements and calculations for one FLIP fuel element and for some standard fuel elements with 12% uranium concentration. There are more possible reasons for these discrepancies. Different FLIP fuel elements contain different erbium concentration (see Table 2) and these differences were not accounted in our measurements. These differences could be eliminated normalizing measured reactivity values to some "nominal" erbium concentration [9]. Our additional analyses [10] showed, that reactivity method for burnup determination is not very appropriate for burnup determination of fuel elements containing strong burnable poisons and for fuel elements with relatively small burnup (e.g. standard fuel elements with 12% uranium concentration in this case). Origins of the problem are variations of fuel element compositions in the manufacturer's acceptable range. Even small variations in H/Zr ratio or in erbium concentration have significant influence on measured reactivity and can screen burnup effect on reactivity.

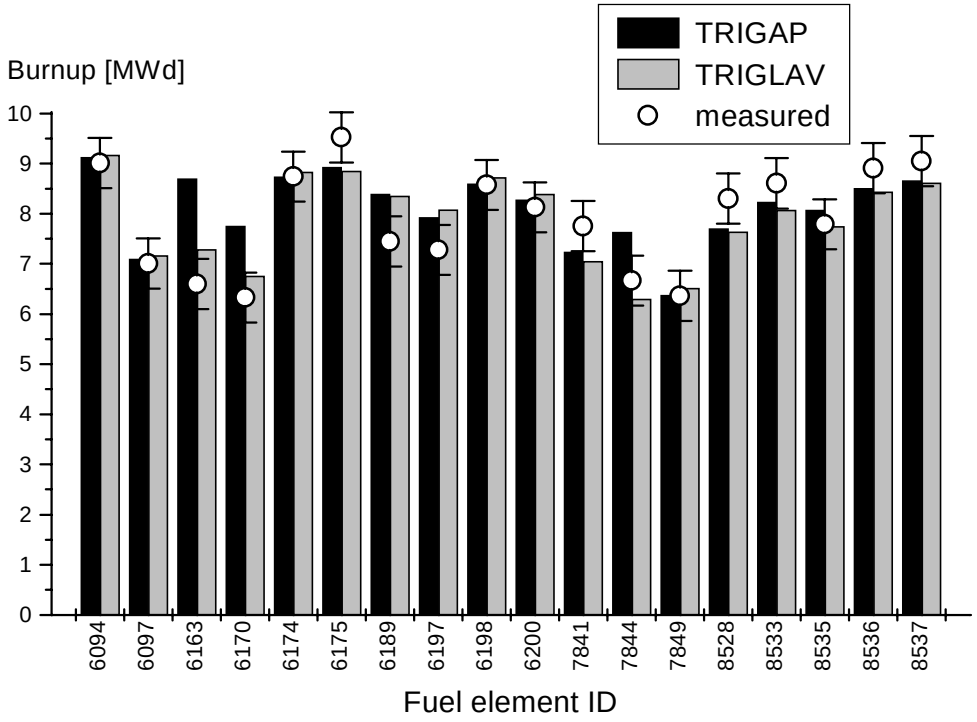

Figure 5. Calculated and measured burnup of selected standard 8.5% uranium concentration, 20% enriched fuel elements.

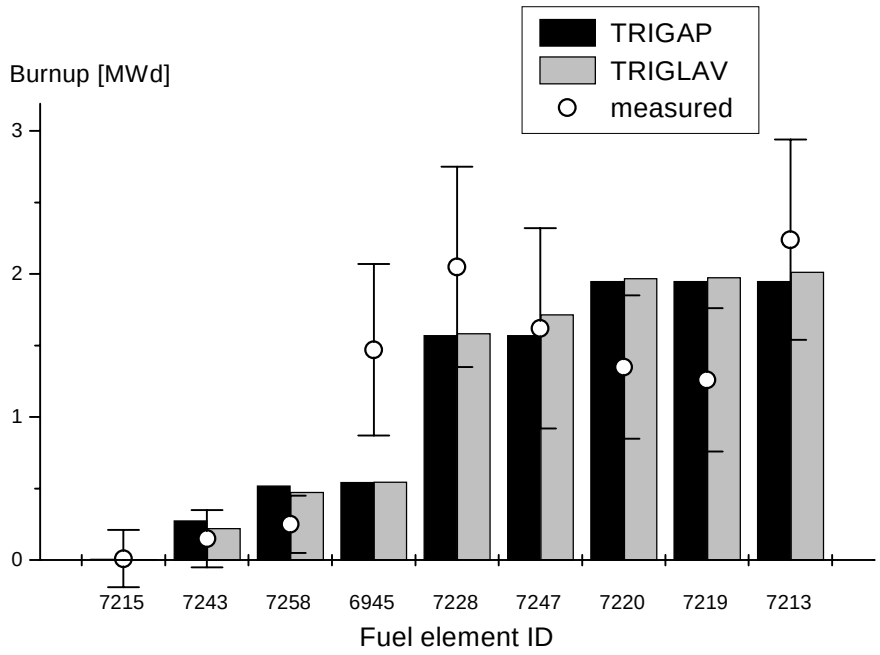

Figure 6. Calculated and measured burnup of selected standard 12% uranium concentration, 20% enriched fuel elements.

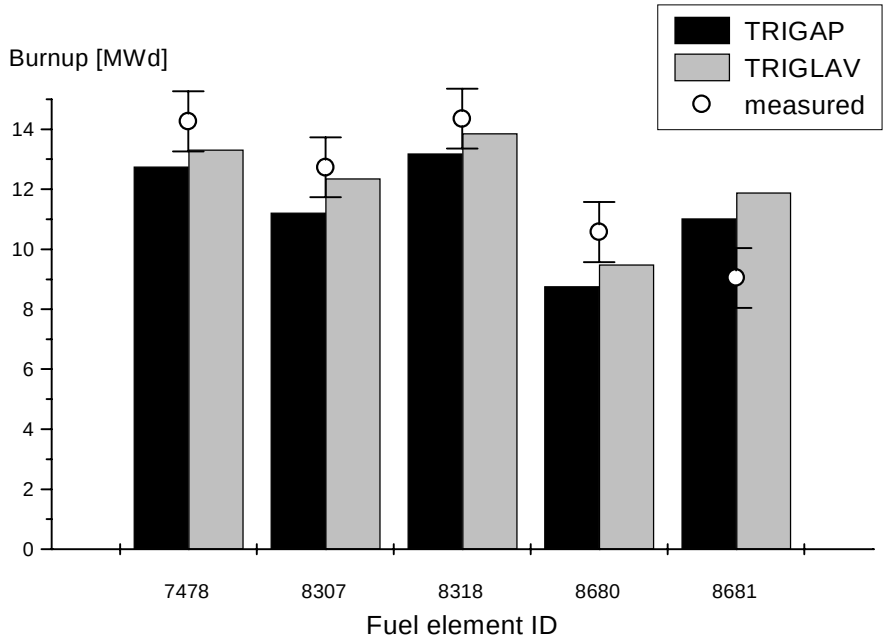

Figure 7. Calculated and measured burnup of selected FLIP 8.5% uranium concentration, 70% enriched fuel elements.

Conclusions

The burnup of TRIGA fuel elements was investigated by reactivity measurements and by reactor calculations with TRIGAP and TRIGLAV code. From the analysis presented in this paper we may conclude that fuel element burnup estimates can be significantly wrong if one uses calculation models that are too simplified, particularly in mixed cores. The analysis presented in this paper applies to TRIGA reactor with mixed LEU - HEU core. Burnup estimates for fuel element in mixed LEU - HEU ring calculated with simple one-dimensional code can be wrong up to 50% or even more. We also performed the analysis for mixed cores made only of LEU fuel elements differing in uranium concentration (e.g., 8.5% and 12% uranium concentrations). The same general conclusions may be drawn as for the combination of HEU and LEU fuel elements except that the effects are not to strong.

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