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Simulation of the TRIGA Mark II Benchmark Experiment with Burned Fuel

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Abstract: A criticality experiment with burned fuel elements was performed in 1998 at "Jožef Stefan" Institute TRIGA Mark-II reactor, with the same fuel elements in the same loading pattern as the TRIGA benchmark experiment performed in 1991. The fuel elements were 2.8% burned on average. The main aim of this work was to test two computer codes, TRIGLAV and MCNP, against this experiment. TRIGLAV is an in-house developed deterministic code, based on diffusion approximation of transport equation, which uses WIMSD program to calculate a unit-cell averaged cross section data. MCNP is a continuous energy 3D Monte Carlo code. Burned fuel isotopic composition was calculated using WIMSD code. Criticality calculations were performed with MCNP and TRIGLAV codes. Burnup reactivity reduction of approximately 1.5% Δk/k was calculated, indicating discrepancy of approximately 0.3% Δk/k to the measured value. This over-prediction can be explained with the observed underestimation of the reactor thermal power due to calorimetric calibration errors.

1. Introduction

One of the problems encountered when developing computer codes for research reactor design and core management calculations is lack of well-documented experimental test cases for their testing. This problem is especially notable for core management calculations, since there exist almost none benchmark experiments with well-defined fuel element burnup. The TRIGA Mark II benchmark experiment with burned fuel presented in this paper partially fills this experimental data gap.

The first benchmark experiment was performed in 1991 with completely fresh fuel. Several reactor parameters (multiplication factor, excess reactivity, fuel temperature reactivity coefficient, fuel element reactivity worth distribution, radial and axial flux distributions, etc.) were measured at well controlled conditions for two core configurations of which one was critical. The results have been extensively analysed and published [1][2].

The second experiment was performed in 1998 using the same fuel elements and the same loading patterns as in the first one. The average burnup of the fuel elements was approx. 2.8%. The same parameters as in 1991 were tried to be measured. However, this was not entirely possible, as the 1991 critical core configuration turned out to be approx. 2% Δk/k subcritical in the 1998 experiment due to burnup. On the other hand, one of the core configurations, which had approx. 2% Δk/k excess reactivity in 1991, became critical in the 1998 experiment. Results of the 1998 experiment with partly burned fuel are complementary to the results of the experiments of the fresh fuel and may be used for testing the criticality and fuel management calculations on burnup. However they can not be treated as a stand-alone criticality benchmark test case since the burnup and the isotopic composition were not determined experimentally but by using computer codes TRIGLAV [3] and WIMSD [4].

2. Description of the experimental core configurations

Jožef Stefan Institute reactor is a 250 kW TRIGA Mark-II type with annular rings. Detailed description of the experimental facility can be found in [2]. The core, surrounded by thick graphite reflector is at the bottom of a reactor tank filled with water. Schematic top and side views of the core are presented in Figure 1. A schematic drawing of the fuel element is presented in Figure 2. Standard commercially available TRIGA fuel elements containing 12 wt% uranium of 20 wt% enrichment with stainless-steel cladding were used. Dimensions and composition of the fuel elements are provided in Table 1. Four control rods were used in the experiments. Three of them were fuel-follower type with the fuel part practically identical in geometry and composition with the fuel elements. One control rod was an air-follower type. This was also the only irregularity in the core composition when the control rods were completely withdrawn (i.e. in the critical core configuration) as no other inserts or water gaps were present.

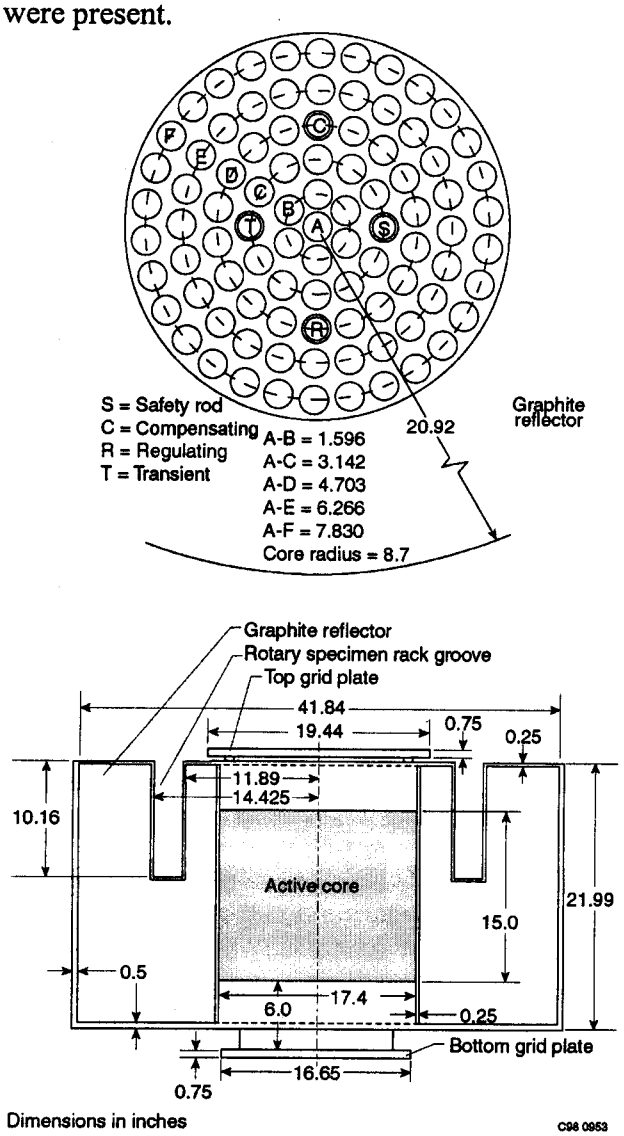

Figure 1. Schematic top and side views of the reactor core and graphite reflector.

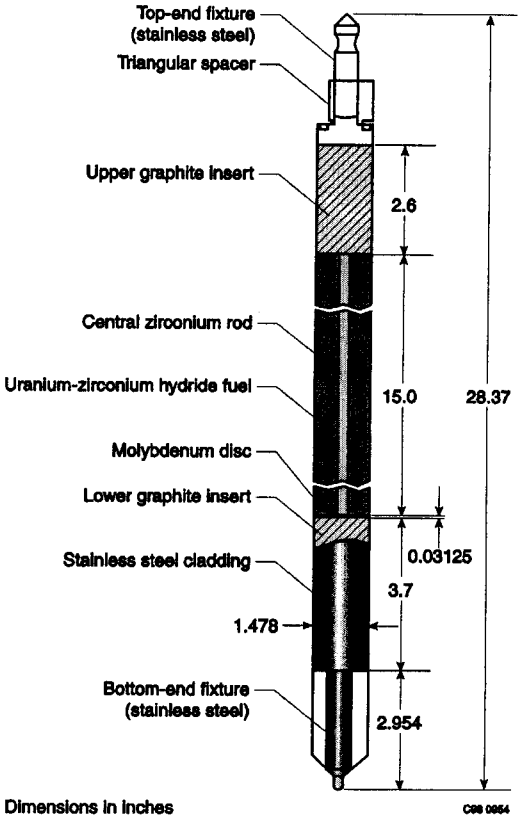

Figure 2. Fuel element.

Table 1. Fuel element basic dimensions and material composition.

Dimensions		Materials	
Outside diameter [cm]	3.754	Fuel meat (U-ZrH) [g]	2300
Fuel meat height [cm]	38.10	U concentration [wt%]	12
Fuel meat diameter [cm]	3.691	U enrichment [%]	20
Diameter of zirconium rod [cm]	0.635	Central rod	Natural Zr
Thickness of cladding [cm]	0.051	Cladding	Type 304 stainless steel

Experimental conditions in the criticality measurements were similar in both experiments and carefully controlled. Reactor temperature was 20°C and uniform across the core, water contained no neutronically important admixtures, gamma background was low, reactor was xenon free, the measurements were performed at negligible thermal power (<100W).

The core excess reactivity was measured in three ways, depending on its range. In the range between 0 and 300 pcm (1 pcm ≡ 10^{-5} Δk/k) it was measured directly from the reactor period using an automatic digital reactivity meter [5]. The error of this measurement was very small (+/- 15 pcm). If the excess reactivity was higher than 300 pcm it was compensated for by inserting a control rod and determined from its calibration curve. Control rod curves were measured by the standard rod exchange method. The error of this measurement was larger (10% relative to the measured excess reactivity) due to the control rod screening effects, redistribution effects in the neutron signal induced by insertion of the control rods and inaccuracy in vertical positioning of the control rods. Negative excess reactivity was determined by the negative reactivity insertion method. One control rod of known reactivity worth was inserted into a subcritical reactor. Multiplication factor was determined from the relative change of the asymptotic flux signal. Two independent detectors were used to eliminate the redistribution effects. Error of this measurement was approximately equal to the case of large positive excess reactivity, i.e. 10% relative, since it originates from the same effects.

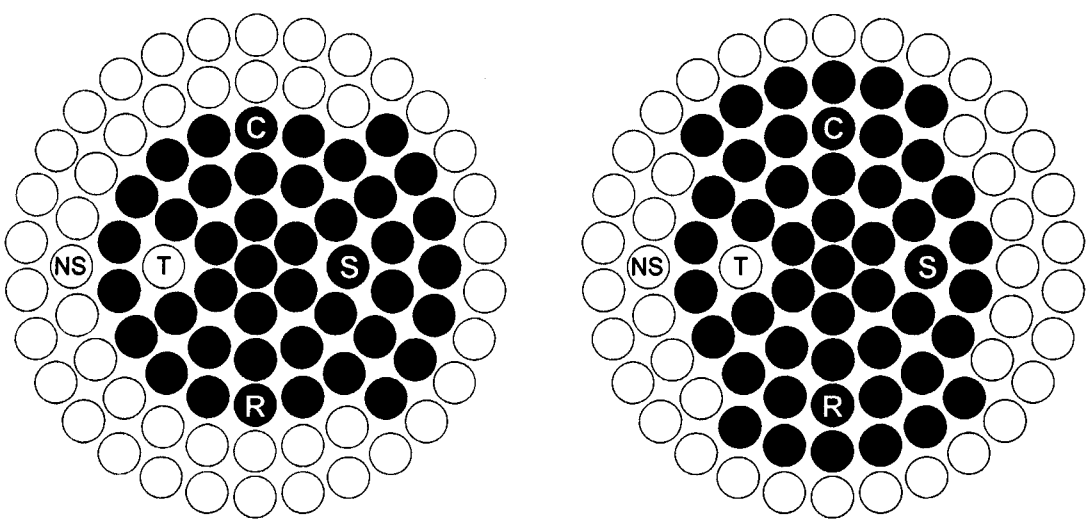

Figure 3. Schematic diagrams of the core 133 (left) and the core 134 (right). Black circles denote fuel element positions. S, C and R in black circles denote fueled-follower control rod positions (safety, compensating and regulating, respectively). T indicates position of the transient control rod. NS is the neutron source position.

Two core configurations, denoted 133 and 134, were treated in both experiments. They are schematically presented in Figure 3. Their multiplication factors are presented in Table 2 together with the estimated inaccuracy of the measurement discussed above. Average calculated fuel element burnup was 1.21 MWd in core 133 and 1.13 MWd in core 134, corresponding to 2.77% and 2.59% burned ^{235}U, respectively. Reactivity reduction rates per burnup unit agree well within relative error interval ±0.1 of excess reactivity measurement for both core configurations. However, the inaccuracy may be larger due to errors in burnup determination.

Table 2. Measured multiplication factor for the cores 133 and 134 loaded with fresh fuel (1991 benchmark experiment [2]) and partly burned fuel (1999 experiment).

Core	Average burnup [MWd]	Average burnup [% burned ^{235}U]	k_{eff}
133	0	0	1.00277 ±0.00015
134	0	0	1.0202 ±0.002
133	1.21	2.77	0.9842 ±0.0011
134	1.13	2.59	1.00460 ±0.00015

3. Fuel element burnup calculation

The fuel element burnup accumulated during 1991-1998 was calculated using the TRIGLAV code [3]. The code is based on a four-group diffusion equation for r-φ geometry, solved by finite differences method. All fuel elements in the reactor are treated in a unit-cell approximation. The unit-cell-averaged cross sections were calculated with WIMSD [4] code, using two different libraries; old 1981 WIMS library modified for some TRIGA specific nuclides (Er 155 and 157, H in Zirconium hydride), and the new based on ENDF/B-VI NJOY calculation [6]. Unit-cell cross sections in WIMSD program are calculated in a 32-group approximation and than averaged to two or four-group structure. The cross-section homogenization was done in two ways, either with a simple flux and volume weighting or using the Effective Diffusion Homogenization (EDH) method in the XSWOUT program [7]. Use of the new library showed no significant change in the calculated burnup. This showed that old WIMS library was suitable for successful prediction of burned fuel element composition. Detailed description of the calculation is presented in [8].

Burnup for all core configurations that operated more than one effective burnup day was explicitly treated. Accuracy of the calculated burnup depends mainly on the experimental power calibration accuracy and precision of the operation records. Total relative calculation error is of the order 10%.

Burnup of selected fuel elements was measured by the reactivity method [9] as a part of the 1998 experiments. The reactivity method is based on the assumption that the reactivity worth of the fuel element is a known, usually linear function of burnup. The measurements were performed in a redesigned reactor core because the fuel elements selected for the measurement had to be removed. The configuration was practically critical. The control rods were withdrawn during the reactivity measurements to eliminate the redistribution effect of the control rods. The digital reactivity meter was used to measure reactivity. Nine fuel elements included in the cores 133 and 134 were selected for the measurements. We started with the element with the highest burnup value. At the end of the measurements this element was measured again to determine reproducibility of the reactivity measurements. The measured burnup values are presented in comparison to the calculated values in Figure 4. Comparison shows agreement within ±1% burned ^{235}U burnup for the fuel elements 7243 and

7258, which spent most of their operating time in the E ring, and for the fuel elements 7220 and 7219 that spent most of their operating time in the C ring. Discrepancy between the measured and calculated burnup for fuel elements 7228, 7247 and 7213, which were positioned in the D or C rings near the transient control rod air-follower, is probably due to the poor fuel element surrounding description in the averaged-unit-cell cross section calculation. In the unit-cell cross section calculation each fuel element is surrounded with six identical fuel elements, which is not the case for the fuel elements near the air-follower. The large discrepancy in the fuel element 6945 is probably due to its different initial isotopic composition (low H/Zr atom ratio).

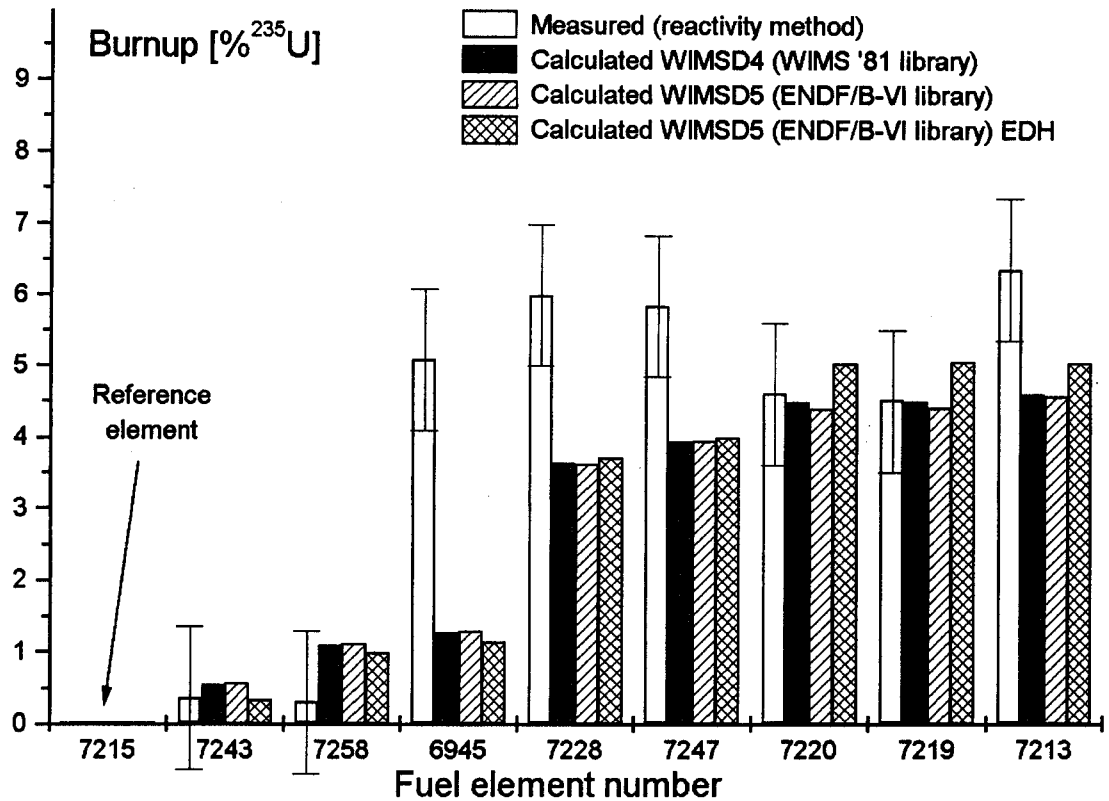

Figure 4. Calculated and measured fuel element burnup.

An important result of the burnup calculation was also the isotopic composition of burned fuel elements. It was calculated with WIMSD for each fuel element at its particular burnup. Isotopic vector is, therefore, limited to the nuclides defined in the WIMS library. 33 fission products are treated explicitly, the rest are considered as a lump pseudo nuclide.

4. Multiplication factor calculations using TRIGLAV

Two-dimensional diffusion codes, such as TRIGLAV, are not appropriate for absolute criticality calculations. Our tests show that a typical difference between a measured and a calculated k_{eff} is about $\pm 2 \cdot 10^{-2}$ if geometrical buckling is used to approximate the leakage in the axial direction. However, if the axial buckling value is experimentally adjusted to a core configuration with a well-defined excess reactivity, agreement between the measured and the calculated k_{eff} will be in the order of $\pm 10^{-3}$ for all similar core configurations and operating conditions. Relative changes in the reactivity between different core configurations can be

calculated more accurately and in this sense the code can be used also for rough criticality predictions.

The TRIGLAV geometry model represents the full TRIGA cylindrical core and graphite reflector. On the reflector outer boundary the zero flux boundary condition is imposed. Every location in the core either occupied by a fuel element, control rod, irradiation channel, or left empty is treated explicitly as a homogeneous region equivalent to the unit-cell. As mentioned above, the unit-cell group constants were calculated with the WIMSD program using old WIMS '81 library, which is integrated in the original TRIGLAV package. The unit-cell homogenisation was done with the simple flux and volume weighting.

Note that in the TRIGLAV simulation the axial buckling factor was adjusted to the experimental core configuration 133 with 43 fuel rods and that the same axial buckling value was used in other calculations. For this reason, the initial fresh core results presented in Table 3 for the examined cores (core configurations 133 and 134) match reasonably well. For burned cores, TRIGLAV over-predicts k_{eff}. The MCNP results show the same level of disagreement for the burned cores. This over-prediction agrees with the observed underestimation of the reactor thermal power. The analysis of TRIGA reactor thermal power calibration method, also presented at this conference [10], shows that the error in calorimetric calibration can be up to 30% in the most unfavourable conditions.

The k_{eff} of burned cores 133 and 134 was calculated twice with the TRIGLAV program. With homogeneous core burnup, where all fuel elements in the core had average core burnup, and with heterogeneous core burnup, where each fuel element had its own burnup. The influence of homogeneous and heterogeneous core burnup distribution on calculation is also presented in Table 3. This calculation showed that the effect of heterogeneous core burnup distribution is minimal and can be neglected in further calculations.

Table 3. Comparison of measured and calculated (MCNP, TRIGLAV) k_{eff} for cores 133 and 134 with fresh and burned fuel.

Core	Burnup	k_{eff}				
	[% ^{235}U]	Experiment	MCNP	TRIGLAV	MCNP - exp.	TRIGLAV - exp.
133	0	1.00277	1.00629	1.00261	$352 \cdot 10^{-5}$	$-16 \cdot 10^{-5}$
	1	-	-	0.99505	-	-
	2	-	-	0.99076	-	-
	2.77	0.9842	0.99105	0.98811	685	391
	2.77[a]	0.9842	-	0.98820	-	400
	5	-	-	0.98139	-	-
	10	-	0.96998	0.96636	-	-
134	0	1.0202	1.02386	1.01885	366	-135
	1	-	-	1.01132	-	-
	2	-	-	1.00698	-	-
	2.59	1.00460	1.00999	1.00485	539	25
	2.59[a]	1.00460	-	1.00456	-	-4
	5	-	-	0.99748	-	-
	10	-	0.98691	0.98225	-	-

[a] Heterogeneous (realistic) core composition.

5. Multiplication factor calculations using MCNP

The MCNP geometry model followed the actual reactor geometry as accurately as possible and reasonable. All fuel elements, control rods, supporting grids, graphite reflector, and water tank were modelled to the highest accuracy. Some simplifications of the geometry

were done by simplifying surroundings of the core, but only to an extent which did not affect k_{eff} significantly. Influence of the simplifications was investigated using a series of MCNP models with different simplification levels compared to the complete model of the TRIGA reactor (which includes a thorough description of the vicinity of the core). The total change in k_{eff} because of the geometry simplifications is an increase in k_{eff} of approximately +150 pcm. For more detailed description of the MCNP geometry model consult [2].

Because not all information on material composition and geometry was perfect, it was necessary to perform extensive sensitivity studies to estimate the effect of these uncertainties on k_{eff}. Sensitivity studies are described in [2] and [11]; here we quote just the main conclusions. Uncertainties in fuel composition data are estimated to be approximately ± 250 pcm. The poorly known composition and fabrication tolerances of the stainless steel clad of fuel elements and control rods are another major source of uncertainty. Combining with other minor sources (e.g. absorber density), all other than fuel material composition data yields approximately ± 500 pcm uncertainty. Combining both uncertainties, the total uncertainty in the calculated k_{eff} for our model is estimated to be ± 560 pcm. It should be noted that even though the geometry and material composition uncertainties are large, they affect only the absolute value of the calculated k_{eff}, while the relative changes (e.g. when calculating effect of burnup) remain unaffected.

The cross section data used for fresh core configurations were taken exclusively from the continuous-energy ENDF/B-VI cross section libraries, except for the thermal cross sections, based on the ENDF/B-IV evaluation. The fission product cross sections were mostly taken from the KIDMAN libraries. Isotopic composition of burned fuel was set up on the basis of the isotopic composition calculated by the WIMSD code. The so-called pseudo fission product, as defined in the WIMS library, was not included in the MCNP burned fuel material specification. However, its effect was determined to be negligible and thus no error was introduced to the model with this simplification.

Results of the MCNP calculations for the fresh and burned cores 133 and 134 are presented in Table 3. The difference between the measured and the calculated k_{eff} is approx. 300 pcm for both fresh cores and grows to approx. 500 and 800 pcm for burned conditions. Consequently, the calculated k_{eff} burnup slope is in both cases approx. 20% smaller than measured, as can be seen in Table 4. This can be explained by the errors and uncertainties in burnup and reactivity measurements and calculations. However, an equal effect in both cores indicates presence of a systematic error, which is most probably due to the power calibration error that accumulated during reactor operation between 1991 and 1999. This assumption is supported by the results of the TRIGLAV code (see Table 4) indicating the same effect. The calculated k_{eff} burnup slope is not dependent on WIMSD library selection in TRIGLAV burnup calculation.

Table 4. MCNP and TRIGLAV calculated and measured Δk/Δburnup for both core configurations.

Core configuration	Experiment [pcm / % burned ^{235}U]	MCNP [pcm / % burned ^{235}U]	TRIGLAV [pcm / % burned ^{235}U]		
133	670	550	520[a]	520[b]	520[c]
134	600	540	550	550	530

Library used:

[a] WIMS '81

[b] ENDF/B-VI

[c] ENDF/B-VI with EDH homogenization method

6. Conclusions

The experiment presented in this paper may be used for testing criticality safety and fuel management codes for burned fuel conditions. Two effects, which can not be easily eliminated, govern uncertainty of the test problem: relatively rough data on initial fuel composition and systematic experimental errors in power calibration during fuel burnup. The first uncertainty could be reduced if better material composition data were released by the fuel manufacturer. The second uncertainty could be reduced by analyzing systematically the thermal power measurements from 1991 to 1998 and by applying corrections. First results of this analysis show that the calculated results for burned cores can be significantly improved by reducing the discrepancy between measured and calculated k_{eff} burnup slope from 20% to less than 10%.

7. References

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