Criticality safety and shielding analysis of WWER-440 fuel configurations

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Abstract

An overview is made of some studies performed on the criticality safety and radiation shielding analysis of irradiated WWER-440 fuel storage and handling configurations. The analytical tools are based on the SCALE 4.4a code system, in combination with the TORT discrete ordinates transport code and the BUGLE-96 cross-sections library. The accuracy of some important results is assessed through comparison with independent evaluations and with measurement data.

Keywords

WWER-440 fuel, criticality safety analysis, shielding analysis, burnup credit

1. Introduction

This paper presents a short overview of studies made in the past several years at the Department of Nuclear Engineering of the University of Sofia on the criticality safety and radiation shielding analysis of spent or irradiated WWER-440 fuel storage and handling configurations. Some of these analyses were performed in response to assignments by the Kozloduy NPP, the Nuclear Regulatory Agency of Bulgaria and the European Atomic Energy Community.

The analytical tools are based on the SCALE 4.4a code system [1], in combination with the TORT discrete ordinates transport code [2] and the BUGLE-96 cross-section library [3]. A number of additional pieces of code have also been created in support of the implemented analytical procedures.

2. Nuclide composition of irradiated WWER-440 fuel

The precision of modelling the nuclide evolution in irradiated fuel is an important prerequisite for the accuracy of subsequent criticality safety and shielding analyses. A related issue is the practicability of reducing the extensive and not easily accountable nuclide inventory in irradiated fuel to a shorter list of important nuclides sufficient for predicting the multiplication properties or the radiation characteristics of typical storage and handling configurations.

The choice of the SCALE code system as a tool for nuclide composition modelling is motivated by its wide availability and by the large validation and verification experience from the application of this system to various fuel designs. Furthermore, many practical problems of nuclide inventory modelling are routinely solved through the direct use of the zero-dimensional ORIGEN-S module of SCALE supplied with pre-generated one-group effective cross-section libraries for a set of standard fuel types.

However, the task of validation and verification of these capabilities of SCALE cannot be considered as fully completed, especially in the case of WWER-type fuel lattices. Thus, the generation of problem-dependent cross-section libraries for ORIGEN-S and the study of the accuracy of WWER fuel nuclide inventory modelling is still a necessary step which is to be performed in conjunction to each particular case of criticality safety or radiation shielding analysis.

Here it should be noted that the cross-section processing methods incorporated in SCALE 4.4a rely on one-dimensional neutron transport models, and this is generally sufficient for the conventional simple WWER-440 fuel assembly design. However, advanced WWER-1000 fuel designs, such as the TVSA with integrated Gd burnable poison fuel pins, may require two-dimensional modelling which is already available in SCALE 5.1 through the discrete ordinates transport module NEWT.

The study of two-dimensional effects in WWER-1000 fuel will not be considered here. Instead, the results and conclusions in this section will be restricted to the application of the SAS2H control module of SCALE 4.4a for the generation of problem-dependent ORIGEN-S cross-section libraries for the simple WWER-440 fuel design types represented in the spent fuel inventories of the Kozloduy NPP.

2.1 Solution of a calculational benchmark of AER

A valuable reference basis for estimating the accuracy of nuclide inventory modelling is the Results Evaluation Report of the CB2 calculational burnup credit benchmark of the Atomic Energy Research community [4].

This portion of the international calculational benchmark on WWER burnup credit has been prepared in collaboration with the OECD/NEA/NSC Burnup Credit Criticality Benchmarks Working Group and proposed to the AER research community. The goal of the study was to compare selected average isotopic concentrations in the fuel pellets of a WWER-440 assembly calculated by the benchmark participants using various fuel depletion codes and libraries. The report contains both the benchmark specifications and the evaluated results.

The comparison results presented in Tables 2.1.1 and 2.1.2 below refer to a WWER-440 assembly with uniform initial enrichment 3.6 %, without integrated burnable poison fuel pins. The burnup values and the corresponding nuclide compositions are average over the fuel assembly. The isotope selection represents a recommended list for burnup credit applications and includes the following nuclides:

- Actinides: U-235, U-236, U-238, Np-237, Pu-238, Pu-239, Pu-240, Pu-241, Pu-242, Am-241, Am-243.
- Fission products: Mo-95, Tc-99, Ru-101, Rh-103, Ag-109, Cs-133, Nd-143, Nd-145, Sm-147, Sm-149, Sm-150, Sm-151, Sm-152, Eu-153, Gd-155.

The nuclide concentration values taken as reference are obtained through averaging the results reported by the benchmark calculation participants, *with the exclusion of those produced using any versions of SCALE or ORIGEN*. Thus, the reference results are based only on the independent codes HELIOS 1.5, CASMO 4, WIMS 7 and 7B, MCU - REA and NESSEL 4 - NUKO.

The reactivity worth of a given nuclide, as given in the comparison tables, represents the relative change, in pcm, of K_∞ due to 1 % relative increase in the number density of this particular nuclide over the reference value. K_∞ is estimated on the basis of the problem-dependent ORIGEN-S effective one-group cross-sections.

The results labelled ‘problem-dependent library’ and ‘PWR 17x17’ library are presently evaluated. The ‘PWR 17x17’ results are obtained using a pre-generated ORIGEN-S library for a 17x17 Westinghouse type PWR fuel assembly. They are closest to the CB2 results among all which can be produced using the pre-generated ORIGEN-S libraries included in SCALE-4.4a. The problem-dependent libraries are based on the 44GROUPNDF5 master cross-sections library of SCALE-4.4a.

The following conclusions can be made:

- The agreement of the results obtained with a problem-dependent ORIGEN-S library with those reported by the CB2 participants (SCALE/ORIGEN results excluded) is generally good and within the observed CB2 uncertainty range.
- The use of a problem-dependent library results in a higher nuclide inventory modelling accuracy, as compared to the best fitting pre-generated PWR fuel library. This is especially true for the major actinides.

A good integral measure of the impact of nuclide composition uncertainties can be their reactivity effect in a selected fuel storage configuration of interest. This approach is illustrated for the case of a compacted wet storage configuration consisting of an infinite triangular lattice with a pitch of 168 mm assembled from hexagonal boronated steel tubes with outer dimension 158 mm, wall thickness 3 mm and boron content 1 wt. %. All tubes contain identical WWER-440 assemblies with the characteristics assumed for CB2. The moderator is pure water with density 0.972 g/cc. Some results, obtained using the KENO-VI Monte Carlo code included in SCALE 4.4a are shown in Table 2.1.3.

2.2 Solution of an AER benchmark based on experimental data

The simplified specification of a calculational benchmark [5] is based on the results of post-irradiation examination of WWER-440 fuel performed under project No. 2670 of the International Science and Technology Centre in Moscow and international partners [6]. The radiochemical assay data are intended for use in spent fuel burnup credit activities.

The comparison results are shown in Table 2.2.1 and refer to eight samples cut out of four fuel pins of a spent fuel assembly from the Novovoronezh NPP with an average burnup of 38.5 MWd/kgU. The decay times are approximately 12.5 years. The presently computed results are obtained using the ORIGEN-S code with problem-dependent libraries generated within the SCALE-4.4a code system.

The larger deviations than in the case of a purely computational benchmark are to be expected. An additional source of error which should be explicitly mentioned is the uncertainty of the assumed burnup values. Nevertheless, with some exceptions, the agreement, at least with respect to the major actinides, is comparable to that observed in the purely computational benchmark.

3. Criticality safety analysis

Criticality calculations of fuel storage and transportation facilities are commonly performed through Monte Carlo simulation because of its higher flexibility in describing various problem configurations, as compared to alternative deterministic methods.

Within the capabilities of SCALE 4.4a this type of analysis can be performed using the KENO-VI Monte Carlo criticality program [7]. The necessary problem-dependent self-shielded multigroup transport cross-section libraries can also be generated within the SCALE system. The nuclide composition of irradiated fuel can be modelled following the approach discussed in Section 2.

The case study presented below refers to the TUK-6 transport cask for 30 WWER-440 fuel assemblies. The cask consists of a basket fitted in a sealed cylindrical steel vessel. The basket is a steel structure comprising a central tube, a rack in which the fuel assemblies are arranged in a regular triangular lattice, and a cylindrical casing. The vessel is filled with water to a certain level above the fuel pin tops. Subcriticality is to be ensured only due to the large inter-assembly pitch, with no fixed or soluble neutron absorbing materials.

The applicable safety regulations for this type of cask require that subcriticality, i.e. $K_{eff} \leq 0.95$, must be guaranteed without soluble boron *at any water density*. Since this is impossible to achieve in the most conservative case of 30 zero-burnup fuel assemblies, the options of reducing the number of stored assemblies or of the introduction of burnup credit have been considered and alternatively applied.

3.1 Comparison with reference results

A reference basis for the presently discussed criticality safety analysis of TUK-6 can be found in the independent reports [8] and [9]. The numerical estimates of K_{eff} in these documents are reported to have been produced using the Russian Monte Carlo code

MMKFK-2. No details about the problem specifications and the cross-section libraries are provided.

The KENO-VI model of the cask includes a detailed heterogeneous representation of the cask components and the fuel pin and assembly lattices. The top and bottom hardware of fuel assemblies is also modelled with an adequate precision through appropriately homogenised material regions. The model layout is shown in Figures 3.1.1 – 3.1.3. The problem-dependent cross-section libraries for KENO-VI are based on the 44GROUPNDF5 master cross-sections library of SCALE-4.4a.

The comparison with reference results is shown in Table 3.1.1 and Figures 3.1.4 – 3.1.6. The reference results for non-zero burnup assume a flat burnup profile and a fuel composition consisting of the following 7 actinides: U-235, U-236, U-238, Pu-239, Pu-240, Pu-241, Pu-242. The decay time is 3 years. The KENO-VI results are obtained under the same assumptions. However, the nuclide concentrations are evaluated independently and do not coincide with the reference ones.

The relative differences in K_{eff} are generally within 1 %, including the non-zero burnup case. (The average absolute relative difference is 1.25 %) With completely different codes and libraries and independently derived problem specifications, these differences can be regarded rather as an estimate of the composite uncertainty bounds of the modelling and calculational procedure.

3.2 Nuclide composition effects

The effects of the more detailed description of the fuel nuclide composition can be examined by solving the same problem with a set of 14 actinides and 15 fission products which is usually employed for burnup credit applications and includes the following nuclides:

- Actinides: U-234, U-235, U-236, U-238, Np-237, Pu-238, Pu-239, Pu-240, Pu-241, Pu-242, Am-241, Am-243, Cm-242, Cm-244.
- Fission products: Mo-95, Tc-99, Ru-101, Rh-103, Ag-109, Cs-133, Nd-143, Nd-145, Sm-147, Sm-149, Sm-150, Sm-151, Sm-152, Eu-153, Gd-155.

In order to assess the adequacy of the above set, it was extended to a list of 18 actinides and 134 fission products, which is a practically complete description of the fuel nuclide composition for the studied range of burnup values and decay times.

The results are shown in Table 3.2.1 and Figures 3.2.1 – 3.2.3. The assumed burnup distribution is flat. The decay time in the 24 MWd/kgU case is 3 years, and in the 36 MWd/kgU case – 3 and 30 years.

It is seen that the actinides only assumption (the 7 actinides listed in Section 3.1) is strongly conservative, while the inclusion of the above presented set of 14 actinides and 15 fission products is sufficient for reproducing almost all burnup-induced reactivity effects.

With a view to the potentially large uncertainties in predicting the concentrations of some important fission products, as illustrated in Section 2, the actinides only approach seems more reliable for burnup credit applications.

The reactivity effect of cooling time, at least up to 30 years, is large and negative, and is due almost entirely to the decay of Pu-241 ($T_{1/2} = 14.34y$) to Am-241 ($T_{1/2} = 432.45y$).

3.3 Axial burnup profile effect

The reactivity effect of the flat axial burnup shape assumption depends in a complicated way both on the actual burnup shapes and on the fuel storage conditions, and has to be assessed separately in each individual case. It must be noted that this reactivity effect may have a strong dependence on the nuclide inventory selection, as well as on the decay time.

While the same considerations should apply to the radial distribution of burnup between the assemblies, the inventory of fuel assemblies in spent fuel configurations is usually quite homogeneous and does not require this type of analysis.

The axial burnup shape effect (also known as "end effect') is illustrated in Table 3.3.1 and Figures 3.3.2 – 3.3.3 below for a representative loading of TUK-6 with 30 spent fuel assemblies with average burnup 36 MWd/kgU. The axial burnup profile is shown in Figure 3.3.1. For the purpose of comparison, the same profile is applied to an average burnup of 24 MWd/kgU.

It is seen that the end effect in this particular case is almost zero at normal moderator density. With the reduction of moderator density the effect becomes small and negative, in correlation with the increased neutron leakage.

4. Shielding analysis

Neutron and photon shielding is usually modelled through solving fixed-source transport problems. While SCALE 4.4a offers such capabilities based on the MORSE-SGC Monte Carlo code [10], it was preferred to employ the TORT discrete ordinates code [2] and the BUGLE-96 cross-sections library [3]. The reasons for this choice were as follows.

- Although Monte Carlo simulations are successfully applied for the evaluation of selected response functions, e.g. the dose rate, within restricted, usually small, regions of the system, they are not quite efficient in reproducing overall distributions of the flux and associated reaction rates.
- The TORT/BUGLE system, together with an expressly developed set of processing utilities, has been successfully applied for neutron fluence calculations in the reactor pressure vessels and the core internals of the WWER-440 and WWER-1000 reactors of the Kozloduy NPP. The analytical procedures have been extensively verified against the measured activities of products of thermal, intermediate and fast neutron reactions in the surveillance positions and behind the pressure vessels of these WWER reactors.

The source terms are determined in conjunction with the fuel nuclide composition by using ORIGEN-S with problem-dependent cross-section libraries (cf. Section 2). The aggregate neutron and photon spectra are produced directly in the energy group structure of BUGLE-96.

The neutron source term evaluation must involve a procedure for accounting for the subcritical multiplication in the analysed fuel configuration. With some acceptable simplification, this can be done through scaling the spontaneous fission and (α,n) source by a factor equal to $1/\left(1-K_{eff}\right)$, where the effective multiplication factor K_{eff} of the analysed fuel configuration can be evaluated as described in Section 3.

Because of the availability of reference data (safety analysis reports and measurement results), the presently described computational procedure is verified for a particular loading of a TUK-6 shipping cask with 30 WWER-440 fuel assemblies with well characterised properties (burnup distribution and irradiation history). The cask loading pattern is shown in Figure 4.1.

4.1 Photon dose rates

A comparison between presently computed and independently evaluated photon dose rates [11] on the cask surface is presented in Tables 4.1.1 and 4.1.2.

The average computed values are in close correspondence to those declared as maximum in the reference source. Considering that the independent evaluation conditions are not reported in the reference source and, in particular, that the sensitivity of the surface dose rate to the assumed dimensions of the steel structures is extremely high, the achieved agreement can be assessed as satisfactory.

4.2 Neutron fluxes and dose rates

For the purpose of verification, the computed neutron flux values above 10 keV were compared with the results from measurements performed at the Kozloduy NPP. The comparison results are presented in Table 4.2.1.

For the studied TUK-6 loading, the subcritical multiplication scaling factor applied to the neutron source is 2.78. It should be noted here that this scaling approach relies on the assumption that the spectrum and shape of the fixed source (almost entirely from spontaneous fission) match those of the neutron-induced fission source. Although the assumption about the shapes is obviously violated, this type of source adjustment is commonly used and the produced results are sufficiently accurate.

The agreement may be assessed as very good, moreover that the restricted set of measurement locations introduces some bias in the estimates of the average, minimum and maximum flux values. The observed 20 % relative difference between computed and measured flux values is also an indirect indicator of the accuracy of the subcriticality calculations involved in the analytical procedure.

Neutron flux measurements in selected locations on the cask surface are routinely performed in fulfilment of a methodology [12] adopted for the application of burnup credit in the criticality safety analysis of the TUK-6 cask. The average value from these measurements is compared with that predicted by an empirical calibration dependence of the measured flux on enrichment, burnup and decay time. This calibrations relies basically on the steep increase of the Cm-244 content with burnup and on the assumption that the fuel inventory in the cask is fairly homogeneous. Here it must be explicitly noted that this assumption is essential for the validity of the method, because due to the subcritical multiplication the flux measurements may fail to distinguish between certain high burnup loading patterns and mixed high and low burnup ones.

The average and maximum computed values of the neutron dose rate on the cask sidewall are respectively 5 μSv/h and 15 μSv/h, the decay time being 11.66 years. Because of the large distance between the fuel region and the top and bottom surfaces of the cask, as well as due to the water shielding, the neutron dose rates there are negligible.

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Tables and Figures

Table 2.1.1. WWER-440 fuel, 3.6% enrichment, burnup 30 MWd/kgU. Relative deviations of selected nuclide concentrations from the CB2 average (with SCALE/ORIGEN results excluded).

	reactivity worth		CB2 evaluations				SCALE4.4a/ORIGEN-S			
			min		max		problem-dependent library		PWR 17x17 library	
decay time	zero	5 y	zero	5 y	zero	5 y	zero	5 y	zero	5 y
nuclide	pcm/%		%		%		%		%	
U-235	**162**	**164**	**-2.4**	**-1.7**	**1.7**	**1.4**	**-1.0**	**-1.4**	**-4.2**	**-4.5**
U-236	**-9**	**-9**	**-2.0**	**-1.3**	**2.7**	**1.6**	**2.0**	**2.6**	**2.0**	**2.6**
U-238	**-219**	**-224**	**-0.8**	**-0.9**	**0.2**	**0.0**	**-0.8**	**-0.9**	**-0.8**	**-0.9**
Np-237	-4	-4	-8.5	-7.2	6.0	4.8	4.3	6.2	0.6	2.3
Pu-238	-1	-1	-3.8	-3.7	3.6	4.8	1.2	1.1	-5.0	-5.5
Pu-239	**200**	**205**	**-4.4**	**-4.9**	**6.3**	**5.6**	**-3.0**	**-3.5**	**-10.7**	**-11.1**
Pu-240	**-73**	**-74**	**-5.7**	**-5.5**	**5.1**	**3.9**	**2.0**	**0.9**	**0.0**	**-1.1**
Pu-241	**56**	**45**	**-2.6**	**-5.4**	**4.1**	**4.5**	**1.3**	**1.3**	**-3.6**	**-3.1**
Pu-242	-4	-4	-4.7	-4.1	8.8	9.4	7.9	8.5	7.6	8.2
Am-241	-1	-9	-9.1	-5.8	6.3	10.1	4.9	-1.8	-1.9	-6.8
Am-243	-1	-1	-4.8	-4.6	5.8	6.1	18.0	18.3	11.2	11.5
Mo-95	-2	-2	-0.4	-0.5	0.8	0.5	0.8	0.5	1.3	1.0
Tc-99	-5	-5	-1.2	-0.5	1.1	0.5	-0.4	-0.5	0.1	0.0
Ru-101	-2	-2	-1.2	-1.3	0.9	0.5	-1.0	-1.3	-0.7	-1.1
Rh-103	-10	-11	-6.0	-13.8	2.7	4.9	2.7	4.9	2.7	4.4
Ag-109	-2	-2	-17.8	-19.0	15	13.4	23.0	21.2	18.3	17.0
Cs-133	-6	-6	-2.2	-2.1	3.5	3.3	3.8	3.6	3.8	4.5
Nd-143	-9	-9	-2.0	-0.7	2.4	0.3	0.7	0.7	0.1	0.0
Nd-145	-3	-3	-2.5	-3.0	1.8	1.3	0.9	0.4	1.8	1.3
Sm-147	-1	-3	-7.3	-6.9	3.0	8.1	1.8	-0.3	4.6	1.6
Sm-149	-7	-11	-7.5	-8.1	5.4	4.7	-1.3	1.3	-10.8	-5.4
Sm-150	-2	-2	-1.3	-1.5	3.4	3.2	13.7	13.5	14.1	13.9
Sm-151	-4	-4	-37.8	-39.0	16.1	15.6	31.4	30.7	21.8	21.5
Sm-152	-4	-4	-18.4	-18.0	10.4	11.0	18.5	19.1	20.4	21.1
Eu-153	-3	-3	-10.1	-7.2	16.1	5.3	-3.6	0	-4.5	-1.1
Gd-155	-0.04	-2	-32.6	-38.0	74.4	84.8	-35.5	-46.8	-43.0	-48.1

Table 2.1.2. WWER-440 fuel, 3.6% enrichment, burnup 40 MWd/kgU. Relative deviations of selected nuclide concentrations from the CB2 average (with SCALE/ORIGEN results excluded).

	reactivity worth		CB2 evaluations				SCALE4.4a/ORIGEN-S			
			min		max		problem-dependent library		PWR 17x17 library	
decay time	zero	5 y	zero	5 y	zero	5 y	zero	5 y	zero	5 y
nuclide	pcm/%			%		%		%		%
U-235	**123**	**127**	**-4.7**	**-2.6**	**3.1**	**2.6**	**-2.1**	**-2.6**	**-7.3**	**-7.7**
U-236	**-9**	**-10**	**-1.9**	**-1.9**	**2.9**	**1.0**	**1.9**	**1.9**	**1.9**	**1.9**
U-238	**-212**	**-216**	**-0.9**	**-0.9**	**0.0**	**0.0**	**-0.9**	**-0.9**	**-0.9**	**-0.9**
Np-237	-6	-6	-9.0	-8.1	5.7	4.9	4.9	6.5	1.6	3.3
Pu-238	-2	-2	-3.9	-3.7	4.3	4.5	1.6	0.7	-4.7	-5.2
Pu-239	**241**	**252**	**-5.4**	**-6.6**	**7.4**	**5.9**	**-2.0**	**-3.3**	**-11.5**	**-12.5**
Pu-240	**-82**	**-83**	**-7.3**	**-6.5**	**5.9**	**4.6**	**2.4**	**1.0**	**-0.3**	**-1.4**
Pu-241	**82**	**66**	**-5.1**	**-6.4**	**5.1**	**5.1**	**0.0**	**-0.3**	**-6.7**	**-6.4**
Pu-242	-6	-6	-4.9	-4.3	9.2	9.9	7.0	7.8	8.5	9.2
Am-241	-2	-12	-10.5	-6.9	8.3	9.3	5.3	-3.1	-3.0	-9.5
Am-243	-2	-2	-5.8	-5.0	4.3	4.3	15.1	16.1	12.9	14.0
Mo-95	-3	-3	-0.4	-0.6	0.7	0.6	0.7	0.4	1.3	1.2
Tc-99	-6	-6	-1.4	-0.6	1.2	0.8	-0.6	-0.8	-0.2	0.0
Ru-101	-2	-2	-1.0	-1.4	1.0	0.6	-1.2	-1.6	-0.6	-1.0
Rh-103	-12	-13	-8.8	-15.2	3.1	5.0	3.1	4.6	2.7	3.9
Ag-109	-3	-3	-18.6	-19.8	15.2	13.4	22.4	20.5	16.8	15.0
Cs-133	-8	-8	-2.3	-2.6	4.5	4.3	4.0	3.7	4.7	4.5
Nd-143	-10	-11	-2.5	-0.8	2.3	0.8	0.3	0.3	-1.4	-1.4
Nd-145	-4	-4	-2.7	-3.4	2.4	1.7	1.4	0.7	2.1	1.4
Sm-147	-1	-3	-7.8	-7.8	3.4	8.7	1.4	-0.6	4.6	2.1
Sm-149	-7	-11	-7.9	-7.3	5.0	2.7	-1.0	1.3	-11.9	-6.0
Sm-150	-3	-3	-1.6	-2.4	4.0	3.2	15.2	14.3	16.0	15.1
Sm-151	-5	-5	-45.9	-46.6	19.1	19.3	34.0	34.2	22.7	23.1
Sm-152	-5	-5	-26.7	-25.1	12.3	14.7	23.2	25.8	25.7	28.4
Eu-153	-4	-4	-15.3	-11.9	19.0	6.4	-0.8	3.3	-1.3	2.5
Gd-155	-0.06	-3	-28.7	-39.4	69.1	87.3	-29.8	-47.0	-39.5	-47.8

Table 2.1.3. WWER-440 fuel, 3.6% enrichment, burnup 30 MWd/kgU, decay time 5 years. Effective multiplication factors of a compacted wet storage configuration. The fuel nuclides selection is as shown in Tables 2.1.1 – 2.1.2.

fuel composition	k_{eff}	σ	reactivity effect, pcm
CB2	0.6335	0.0012	--
ORIGEN-S, problem dependent library	0.6234	0.0013	-1594
ORIGEN-S, PWR 17x17 library	0.6076	0.0012	-4088

Table 2.2.1. Relative deviations of computed from measured nuclide concentrations in samples from a WWER-440 fuel assembly with enrichment 3.6% and average burnup 38.5 MWd/kgU.

sample No.	21	149	162	79	57	135	182	69
burnup, MWd/kgU	43.86	45.54	47.44	49.94	36.80	30.48	25.71	31.95
nuclide	%	%	%	%	%	%	%	%
u234	-12.7	-15.3	-17.6	57.4	76.2	-13.3	66.7	16.9
u235	**-10.8**	**-9.6**	**-5.4**	**24.7**	**-0.7**	**-8.3**	**-10.3**	**-2.3**
u236	**7.9**	**6.6**	**5.6**	**8.6**	**10.5**	**13.8**	**15.3**	**9.2**
u238	**-0.2**	**-0.5**	**-0.5**	**-0.8**	**-0.3**	**-0.2**	**-0.3**	**-0.1**
np237	-50.1	-49.4	-48.5	-49.5	-45.7	-39.4	-45.6	-49.8
pu238	38.6	58.7	39.1	28.3	50.0	76.3	48.0	29.5
pu239	**-5.5**	**2.3**	**6.5**	**11.1**	**5.7**	**10.4**	**-2.4**	**-3.2**
pu240	**-4.7**	**-1.7**	**-1.4**	**-1.8**	**2.6**	**7.0**	**6.9**	**0.3**
pu241	**-7.3**	**-1.9**	**0.1**	**4.9**	**1.6**	**5.6**	**7.2**	**-3.6**
pu242	5.7	6.5	3.1	8.8	2.7	17.2	29.3	1.5
am241	3.4	-0.7	5.6	5.8	6.2	13.1	10.2	0.6
am242m	5.0	-9.4	17.5	34.3	9.3	15.2	-20.7	-4.2
am243	35.0	26.3	27.6	15.4	27.2	64.9	56.3	20.6
cm244	14.2	8.9	-5.7	-26.8	-16.4	15.8	3.8	-2.9
cm245	-27.3	8.2	-31.7	-41.4	16.3	-4.8	-22.4	-41.8
cm246	40.6	63.1	68.4	116.1	--	-18.0	-23.8	-40.2
mo95	6.2	8.6	7.2	6.9	8.1	4.6	17.4	3.8
tc99	5.2	8.3	8.0	9.8	6.2	3.2	16.0	2.7
ru101	3.1	6.1	4.8	5.6	3.9	1.4	13.4	1.6
pd105	1.1	1.6	0.0	8.3	-1.8	-4.3	2.3	-1.1
pd108	44.9	45.8	40.9	57.8	40.9	28.6	38.0	30.3
ag109	129.6	140.7	149.1	186.3	114.6	96.8	95.0	98.5
cs133	10.7	6.6	6.1	10.8	1.4	1.1	10.8	1.8
cs134	-9.6	18.5	-5.7	2.9	-13.3	-12.1	-18.3	-19.2
cs135	-1.1	-2.5	0.2	6.1	2.0	0.0	1.0	-4.3
cs137	8.7	-42.3	4.8	10.1	2.7	2.7	12.4	0.7
ce140	11.0	13.6	12.8	12.2	12.1	9.6	22.6	9.7
ce142	5.9	8.2	6.8	6.0	7.0	5.6	17.8	4.5
nd142	9.5	9.7	12.4	10.1	7.3	-3.9	33.3	25.2
nd143	2.7	4.2	4.0	4.8	5.6	3.7	16.8	5.1
nd144	5.5	7.4	7.1	2.9	2.7	2.8	16.7	1.7
nd145	-5.0	-3.8	-3.5	-4.9	-4.2	-3.3	10.3	-4.0
nd146	14.3	16.3	16.7	14.0	11.2	9.0	22.3	9.0
nd148	2.3	4.1	4.8	5.2	1.7	0.6	13.1	0.0
nd150	4.8	7.3	8.3	12.2	5.9	1.9	14.7	1.8
sm147	2.3	-3.7	0.3	3.8	0.3	-0.6	-8.0	-2.7
sm148	-6.8	-10.1	-9.4	-8.0	-8.4	-9.2	-7.8	-8.8
sm149	33.8	29.5	34.0	43.9	35.5	13.0	2.9	26.8
sm150	30.3	25.1	27.6	32.5	22.5	16.4	11.1	17.5
sm151	76.9	67.0	79.2	92.2	66.3	43.0	29.4	56.3
sm152	44.6	37.9	43.1	49.6	35.7	31.2	20.9	29.2
sm154	19.1	15.5	18.0	22.5	12.7	-64.1	6.4	10.1
eu151	830.0	194.4	498.8	1215.8	366.1	244.2	3.8	181.6
eu153	661.3	727.1	812.2	1209.2	517.5	359.8	305.0	400.0
eu154	639.4	775.0	900.0	1311.8	566.7	417.5	327.5	385.0
eu155	400.0	479.2	669.6	788.3	276.2	215.3	158.1	230.4
gd155	423.1	430.0	453.1	502.8	370.2	306.3	292.0	326.0

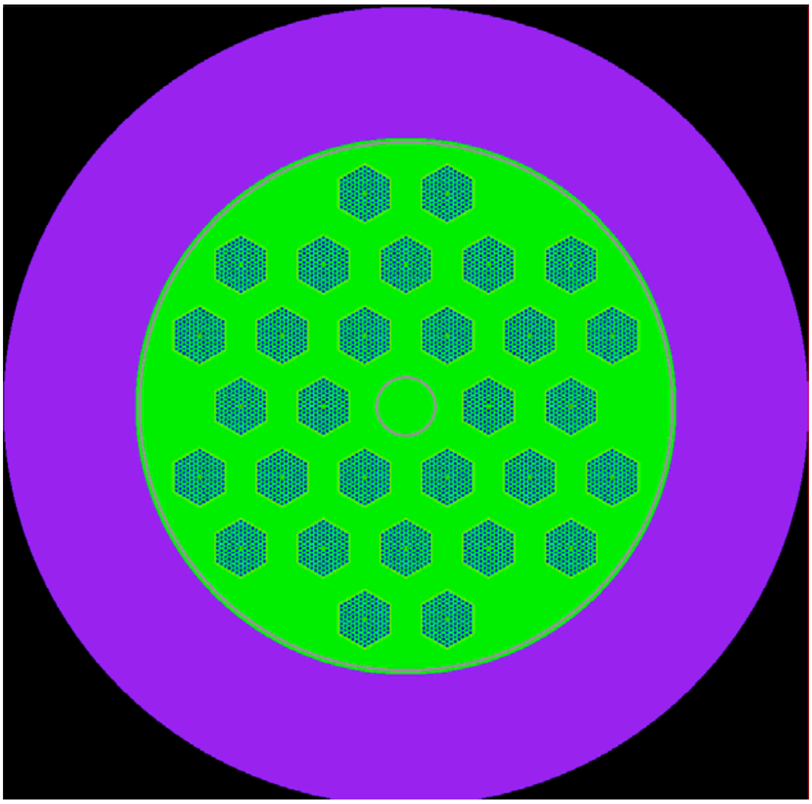

Figure 3.1.1. Horizontal section of a TUK-6 shipping cask with 30 assemblies

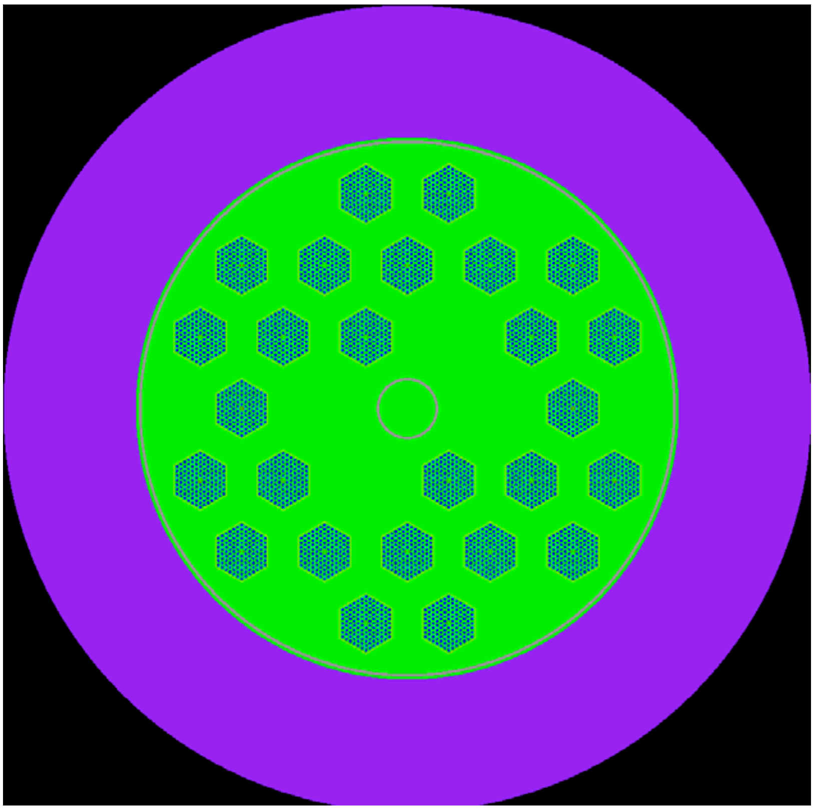

Figure 3.1.2. Horizontal section of a TUK-6 shipping cask with 26 assemblies

Figure 3.1.3. Vertical section of a TUK-6 shipping cask

Table 3.1.1. K_{eff} of TUK-6 with 3.6% WWER-440 fuel assemblies. The reference results are from [8] and [9], the K-VI results are presently evaluated.

water density, g/cm^3	26 FA, BU=0		30 FA, BU=0		30 FA, BU=24 MWd/kgU	
	ref.	K-VI	ref.	K-VI	ref.	K-VI
0.972	0.802	0.817	0.840	0.862	0.780	0.774
0.8	0.829	0.843	0.888	0.904	0.825	0.817
0.6	0.875	0.890	0.962	0.975	0.873	0.884
0.5	0.905	0.913	1.001	1.014	0.914	0.918
0.4	0.926	0.934	1.034	1.043	0.925	0.947
0.3	0.930	0.947	1.046	1.051	0.934	0.950
0.2	0.927	0.934	1.007	1.007	0.895	0.912
0.1	0.820	0.827	0.841	0.848	0.741	0.751

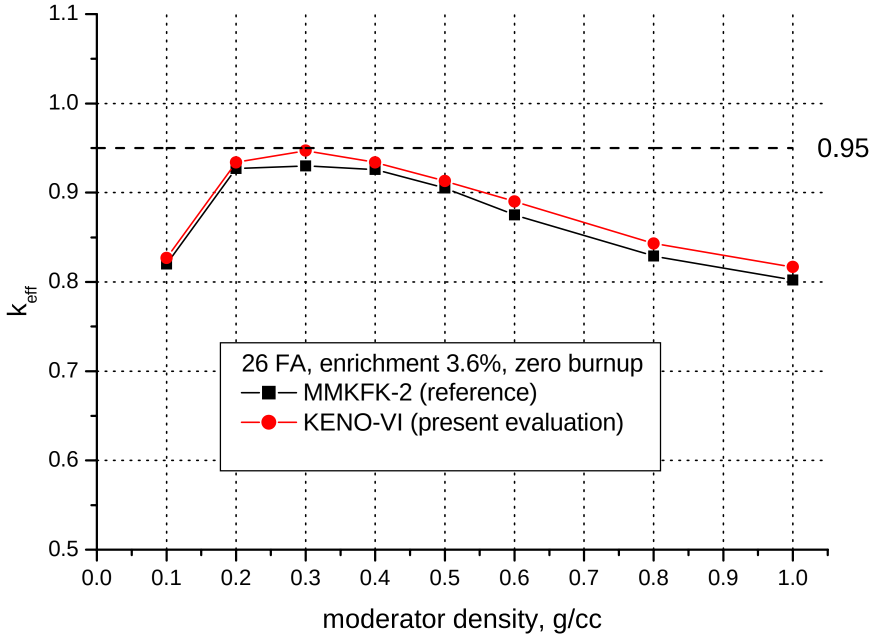

Figure 3.1.4. K_{eff} of TUK-6 with 26 WWER-440 fuel assemblies. Enrichment 3.6%, zero burnup.

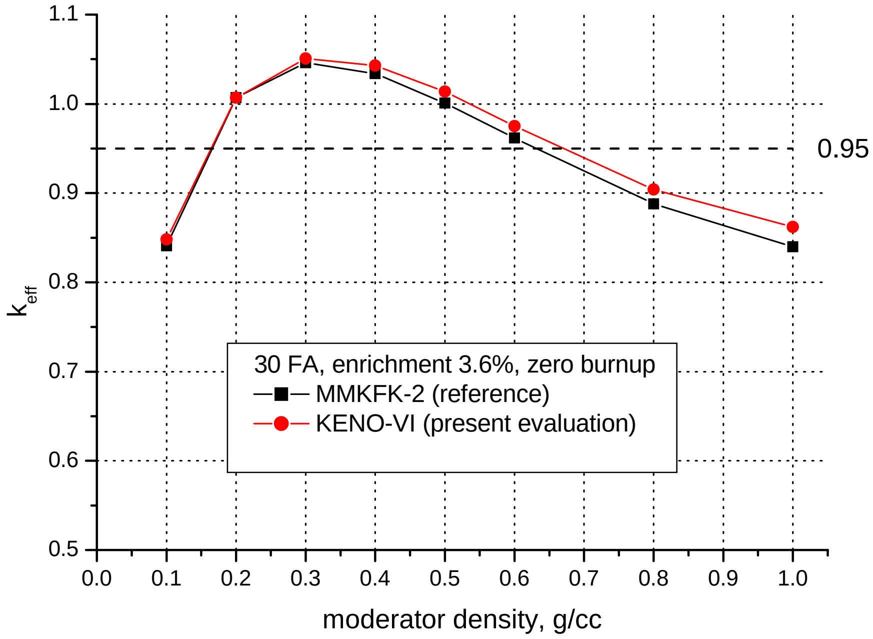

Figure 3.1.5. K_{eff} of TUK-6 with 30 WWER-440 fuel assemblies. Enrichment 3.6%, zero burnup.

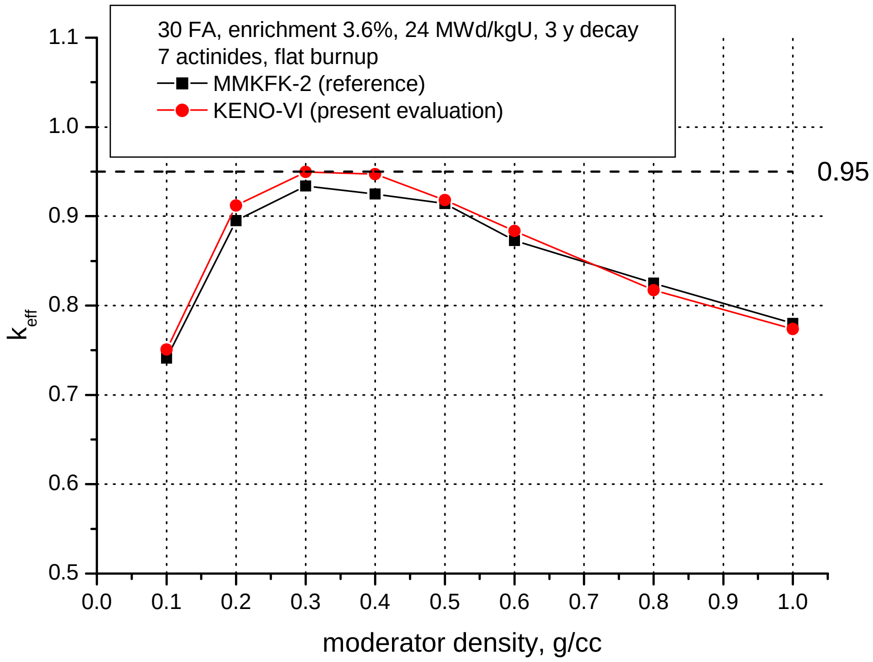

Figure 3.1.6. K_{eff} of TUK-6 with 30 WWER-440 fuel assemblies. Enrichment 3.6%, burnup 24 MWd/kgU. Flat burnup, actinides only. Decay time 3 years.

Table 3.2.1. K_{eff} of TUK-6 with 30 WWER-440 fuel assemblies. Initial enrichment 3.6%. Flat burnup.

burnup	24 MWd/kgU			36 MWd/kgU		
decay time	3 years			3 years		30 years
composition	7 ACT	14 ACT 15 FP	18 ACT 134 FP	14 ACT 15 FP	18 ACT 134 FP	
water density, g/cm^3						
0.972	0.774	0.720	0.711	0.662	0.648	0.605
0.8	0.817	0.758	0.749	0.697	0.680	0.636
0.6	0.884	0.822	0.808	0.752	0.741	0.690
0.5	0.918	0.851	0.839	0.782	0.767	0.718
0.4	0.947	0.875	0.864	0.803	0.789	0.736
0.3	0.950	0.879	0.869	0.809	0.792	0.741
0.2	0.912	0.842	0.827	0.772	0.754	0.705
0.1	0.751	0.696	0.681	0.635	0.620	0.577

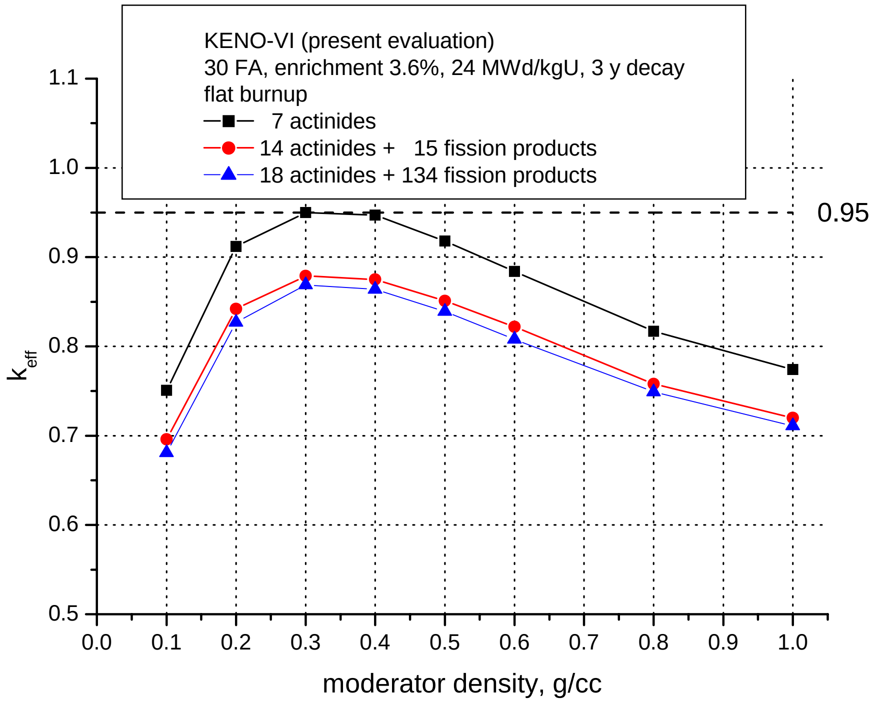

Figure 3.2.1. K_{eff} of TUK-6. Burnup 24 MWd/kgU. Fuel nuclide inventory effects.

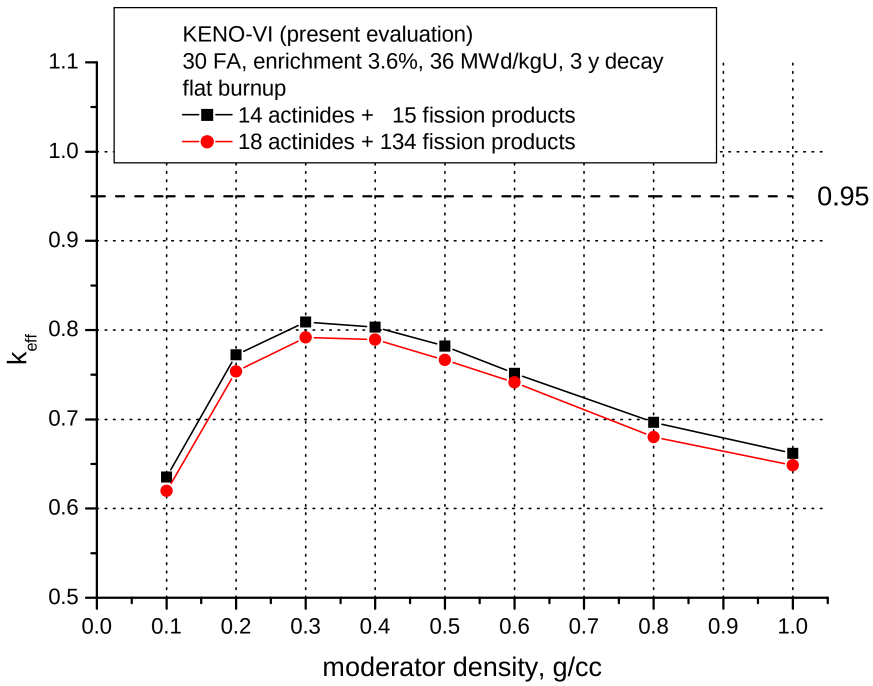

Figure 3.2.2. K_{eff} of TUK-6. Burnup 36 MWd/kgU. Fuel nuclide inventory effects.

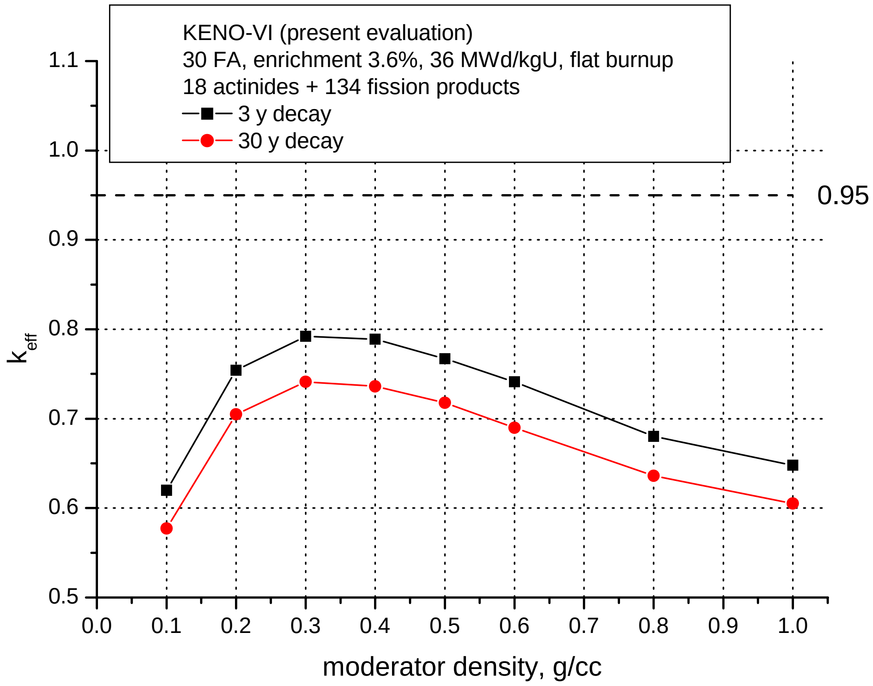

Figure 3.2.3. K_{eff} of TUK-6. Burnup 36 MWd/kgU. Decay time effects.

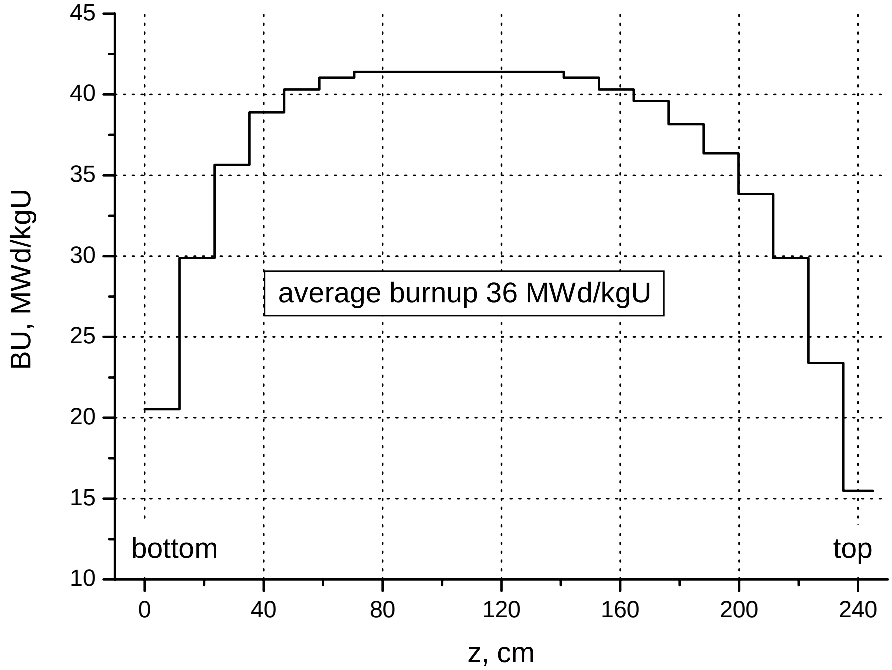

Figure 3.3.1 Axial burnup profile for a realistic loading of TUK-6.

Table 3.3.1. K_{eff} of TUK-6 with 30 WWER-440 fuel assemblies. Initial enrichment 3.6%. Axial burnup shape effect.

burnup	24 MWd/kgU		36 MWd/kgU	
decay time	3 years			
composition	14 ACT 15 FP			
axial shape	flat	profiled	flat	profiled
water density, g/cm^3				
0.972	0.720	0.717	0.662	0.667
0.8	0.758	0.753	0.697	0.696
0.6	0.822	0.811	0.752	0.745
0.5	0.851	0.840	0.782	0.770
0.4	0.875	0.865	0.803	0.793
0.3	0.879	0.866	0.809	0.793
0.2	0.842	0.827	0.772	0.757
0.1	0.696	0.686	0.635	0.627

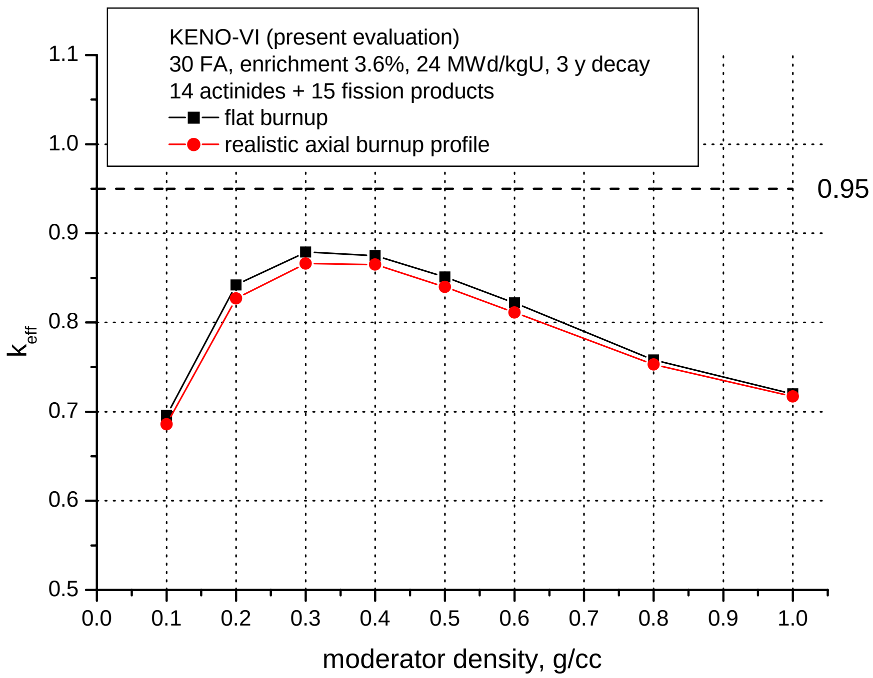

Figure 3.3.2. K_{eff} of TUK-6. Burnup 24 MWd/kgU. End effect.

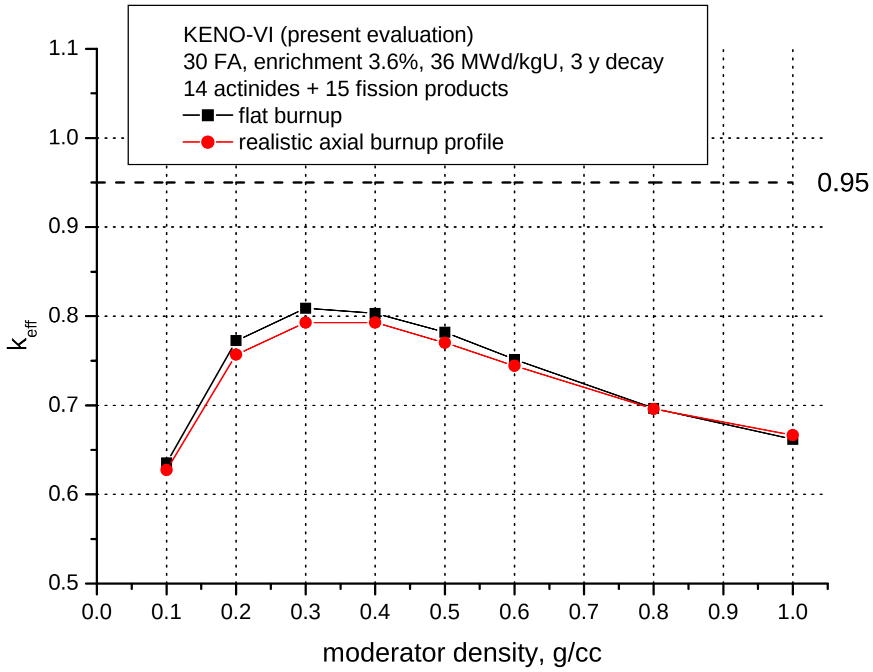

Figure 3.3.3. K_{eff} of TUK-6. Burnup 36 MWd/kgU. End effect.

```
                 BASKET 425

                  30       31
                '32764' '32767'
                 35.417  35.903
      25       26       27       28       29
    '32763' '33052' '33053' '33054' '32820'
     35.461  36.034  35.191  36.471  34.823
  19       20       21       22       23       24
'32762' '33051' '33067' '33068' '33055' '33026'
 34.765  35.990  36.415  35.437  35.119  36.609
      14       15       16       17       18
    '33047' '33066' '         ' '33069' '33058'
     36.036  36.336                 36.347  36.653
   8        9       10       11       12       13
'32761' '33045' '33064' '33063' '33059' '33030'
 34.590  35.175  36.033  36.028  36.450  36.609
       3        4        5        6        7
    '31281' '33044' '33041' '33062' '33035'
     36.863  36.598  35.141  35.413  36.503
                   1        2
                '31279' '33039'
                 38.385  36.619

  1     - assembly position
'31279' - assembly identifier
 38.385 - average burnup, MWd/kgU
```

Figure 4.1. Loading pattern of the TUK-6 cask.

Table 4.1.1. Computed surface photon dose rates for a particular TUK-6 fuel loading. Average burnup 36 MWd/kgU, decay time 3 years.

sidewall	average 26 μGy/h, maximum 63 μGy/h
top	average 16 μGy/h, maximum 62 μGy/h
bottom	average 9 μGy/h, maximum 43 μGy/h

Table 4.1.2. Independently evaluated photon dose rates on the surface of TUK-6. Burnup 40 MWd/kgU, decay time 3 years.

sidewall	maximum 34 μSv/h
top	maximum 18 μSv/h
bottom	maximum 10 μSv/h

Table 4.2.1. Computed and measured neutron flux values above 10 keV on the sidewall of TUK-6. The estimated measurement error is 10 %.

	average	minimum	maximum
computed	227.0 $cm^{-2}s^{-1}$	195.4 $cm^{-2}s^{-1}$	249.4 $cm^{-2}s^{-1}$
measured	187.8 $cm^{-2}s^{-1}$	142.3 $cm^{-2}s^{-1}$	218.8 $cm^{-2}s^{-1}$