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Neutronics calculations for specified cells and power histories

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1. Introduction

For the validation of methods and cross section data for HTR-applications several benchmarks and cell calculations were performed. Especially the influence of different methods for solving the burn-up equations were tested and the influence of nuclear data. Benchmarks from Int. Handbook of Criticality Safety Experiments **[1]** were used for comparisons (graphite moderated systems) as well as UO_2 Pebble fuel assemblies with different U-load (coated particle fuel as in PBMR) and some $^{233}UO_2$-ThO_2 systems were investigated. Burn-up calculations were performed for cells like in the HTR-N1-benchmark exercise.

2. Graphite moderated benchmarks

For graphite moderated systems there are benchmarks from **[1]** available. 18 cases of these benchmarks were calculated with MCNP **[2]** based on JEF-2.2 nuclear data prepared at IKE. The results of the calculations are shown in Fig. 1. The comparison of the calculated k_{eff}-values with the experimental values show a remarkable over-estimation. This is in accordance to other calculations published in **[1]**. The main reasons will be in unsatisfying documentation of the experimental data.

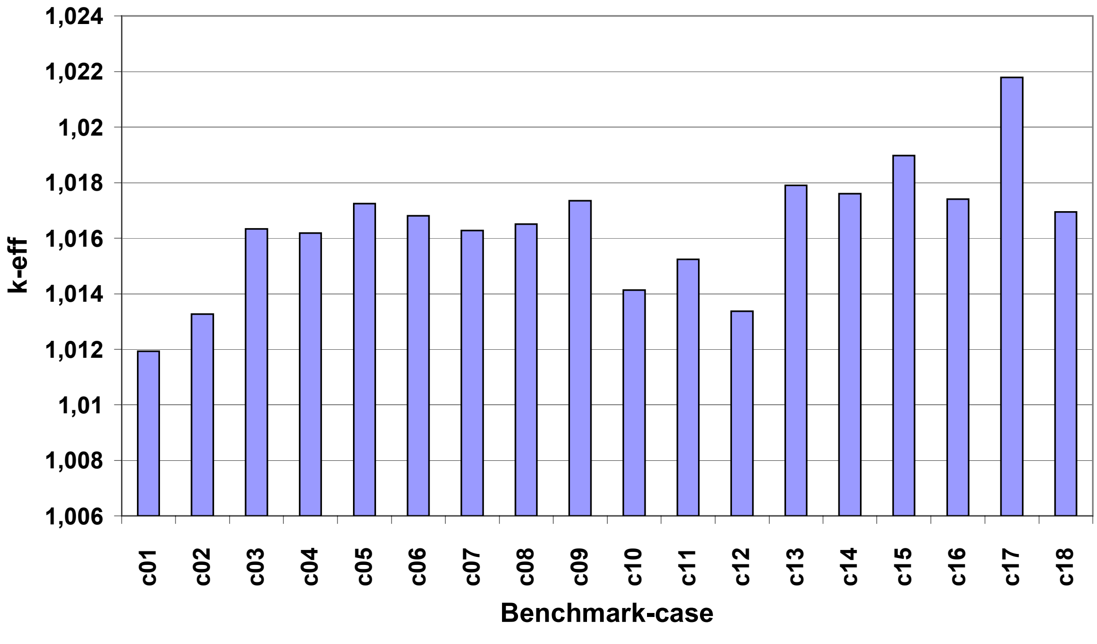

Fig. 1: Benchmark hct-02 with graphite as moderator (from [1]**)**

3. UO_2 fuel cell calculations

For a typical HTR pebble fuel element (specification see **[3]**) as shown in Fig. 2 several calculations with a variation of the U load/pebble were performed with MCNP based on different nuclear data evaluations. The results are shown in Fig. 3. There is a distinct difference for the 3 evaluations, especially the ENDF/B-VI.5 values are remarkable low (see Fig. 4 and Fig. 5). Later calculations, based on ENDF/B-VI.8 or JEFF-3.1 showed good agreement with JEFF-2.2 and JENDL 3.2.

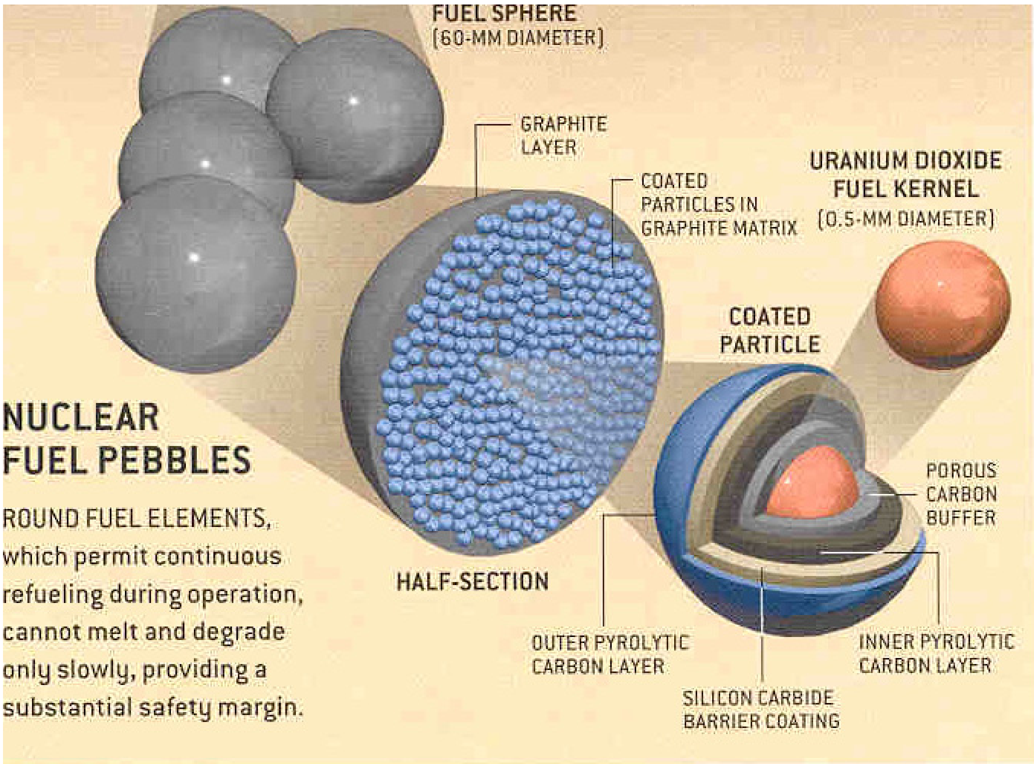

Fig. 2: Nuclear fuel pebbles

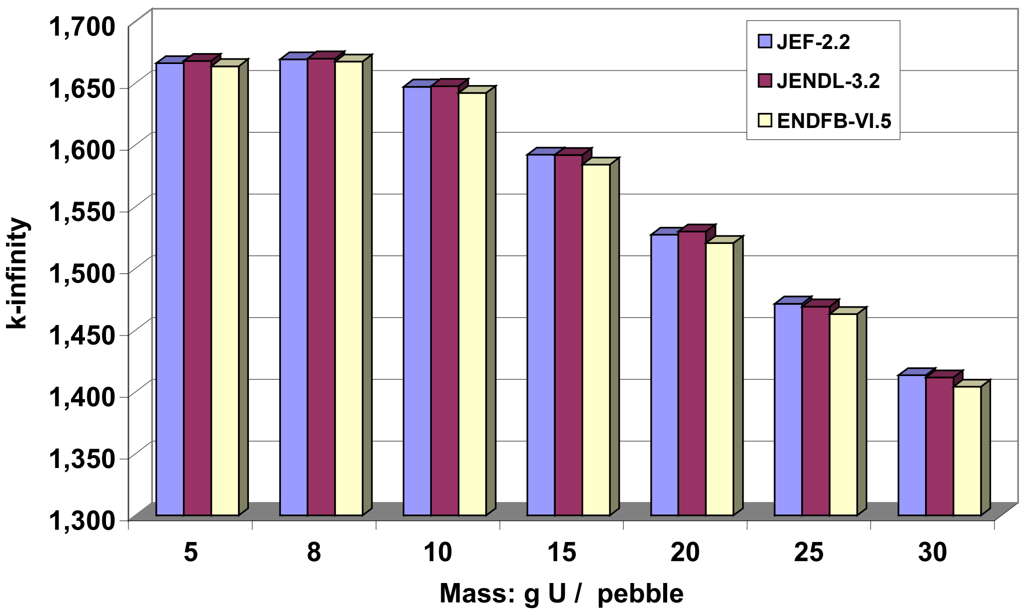

Fig. 3: Comparison of k_inf of different evaluations as a function of U load / HTR pebble

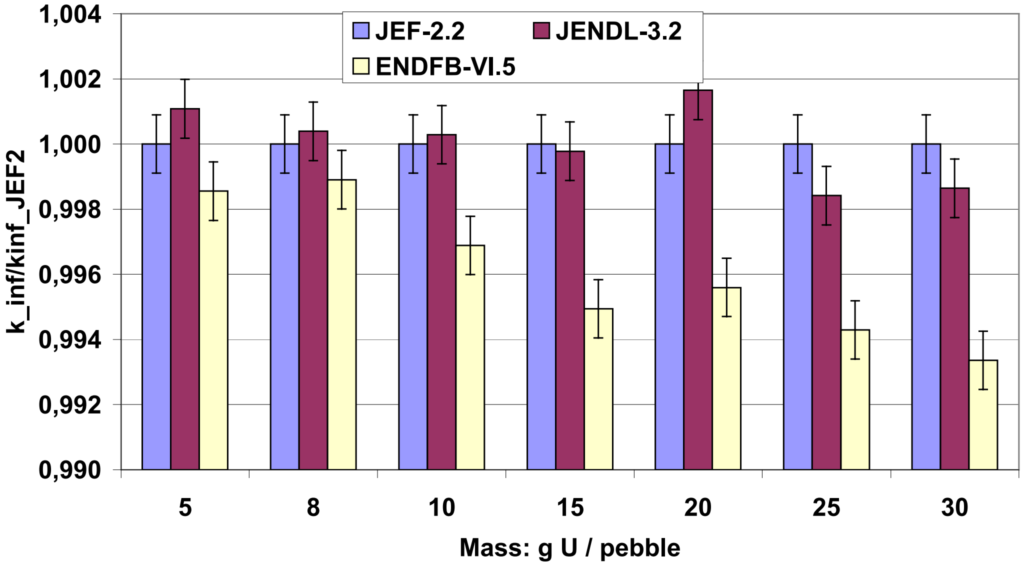

Fig. 4: Comparison of k_inf of different evaluations as a function of U load / HTR pebble. Reference value: kinf_Jef2

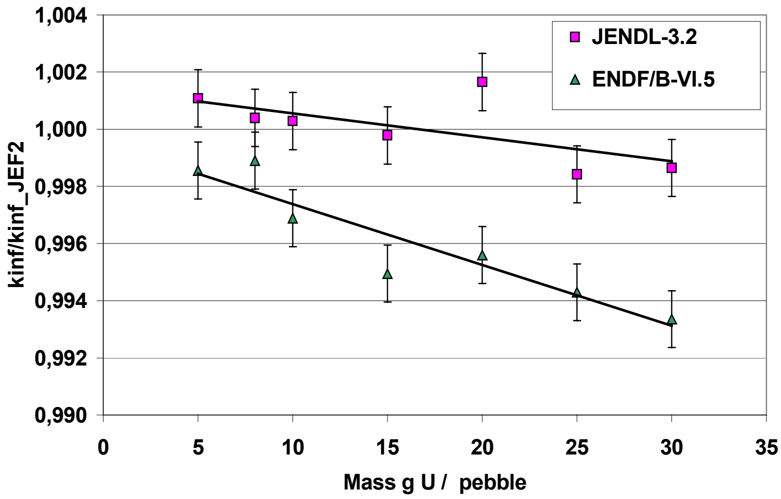

Fig. 5: Comparison of k_inf of different evaluations as a function of U load / HTR pebble. Reference value: kinf_JEF-2.2

4. Infinite $^{233}UO_2$-ThO_2 systems

The neutron multiplication in lattices using $^{233}UO_2$-ThO_2 and highly enriched $^{235}UO_2$ in graphite was studied by recalculating the benchmarks in **[1**]. In Table 1 some of our results are compared with the experimental values. Among the published data we have considered the room temperature and 1023K. For both temperatures the data sets IKE/GA and IKE/ORNL for graphite give identical infinite medium multiplication factors. The agreement with the experimental values is quite good. The differences of 0.25 % for room temperature and 0.06 % for 1023 K may be caused by an unknown content of parasitic absorbers in the graphite and an approximated consideration of the extension of the core structure due to the strong temperature increase.

Table 1 : Experimental and calculated k_∞(T) values for a $^{233}UO_2$-ThO_2 HTGR lattice with C/Th=193 and C/ ^{233}U =10470 for scattering laws using GA frequency spectrum and ORNL frequency spectrum

Temperature in K	Experiment	IKE/GA	IKE/ORNL
293.6	1.0587±0.0014	1.0626	1.0626
1023	1.0245±0.0011	1.0228	1.0229

For a nearly identical ^{235}U-system the calculated k_∞ values also agree completely up to 0.0001 for the two thermal neutron graphite data sets at room temperature as well as at 1023K.

The mentioned calculations were based on different scattering law data derived from GA and ORNL (see **[4**]). From point of view of our comparisons for the neutron scattering cross sections for different data sets show sensitivities of some percent.

Therefore we have not expected large consequences for neutron flux density spectra or reactivities of multiplying systems.

5. Burn-up cell calculations

For the HTR-N1 Pu cell burn-up benchmark coupled neutronics and burnup calculations were performed for 4 different fuel compositions. As neutronics code the Monte Carlo Code MCNP was used, for the depletion calculations the code ORIGEN-2.1 and a standard burnup code based on a reduced nuclide model. The calculations were performed with JEF-2.2.

The benchmark was specified to assess for existing codes the neutronics features and the nuclide compositions for very high burnup up to 800 000 MWd/kg HM. As benchmark geometry a HTR pebble fuel element was chosen with the typical geometric parameters of 6 cm diameter for the sphere and 5 cm diameter of the coated particle zone. The coated particles itself were of a PuO_2 sphere of .024 cm radius and 4 shells with C, graphite, SiC' and Graphite. With the densities 1.05, 1.90, 3.18 and 1.90 g/cm³. The Pu-Oxide density is 10.4 g/cm³. The thickness of the coating layers is 95/40/35 and 40 μm.

The benchmark was calculated with following boundary and power conditions:

- white boundary condition at the pebble surface,
- constant power of 1.0 kW during irradiation,
- target burn-up 800 MWd/kg HM.

The initial composition of the Plutonium for the four benchmark cases are 1st generation plutonium (two alternatives) and 2nd generation plutonium with 1.5 g and 3.0 g Pu respectively.

The benchmark reflects not a realistic environment of sphere with the specified irradiation history, since in a realistic environment the effective multiplication factor is 1.0. The infinite multiplication factor of the specified benchmark system varies however from 1.2 to 0.2. Furthermore, the assumption of a constant power of 1 W requires very high neutron fluxes at high burn-up. A realistic environment is composed of spheres with an average burn-up and constant power of the cluster of spheres with the consequence of decreasing power in the “benchmark sphere”.

The results of the benchmarks are shown in for 1st generation plutonium version A, in for 1st generation plutonium version B, in Fig. 7 for 2nd generation plutonium (1,5 g Pu) and finally in Fig. 8 for 2nd generation plutonium with 3g Pu/pebble. The calculations were made with a coupled system MCNP and a standard burn-up code with 105 Nuclides and ORIGEN with 300 Nuclides.

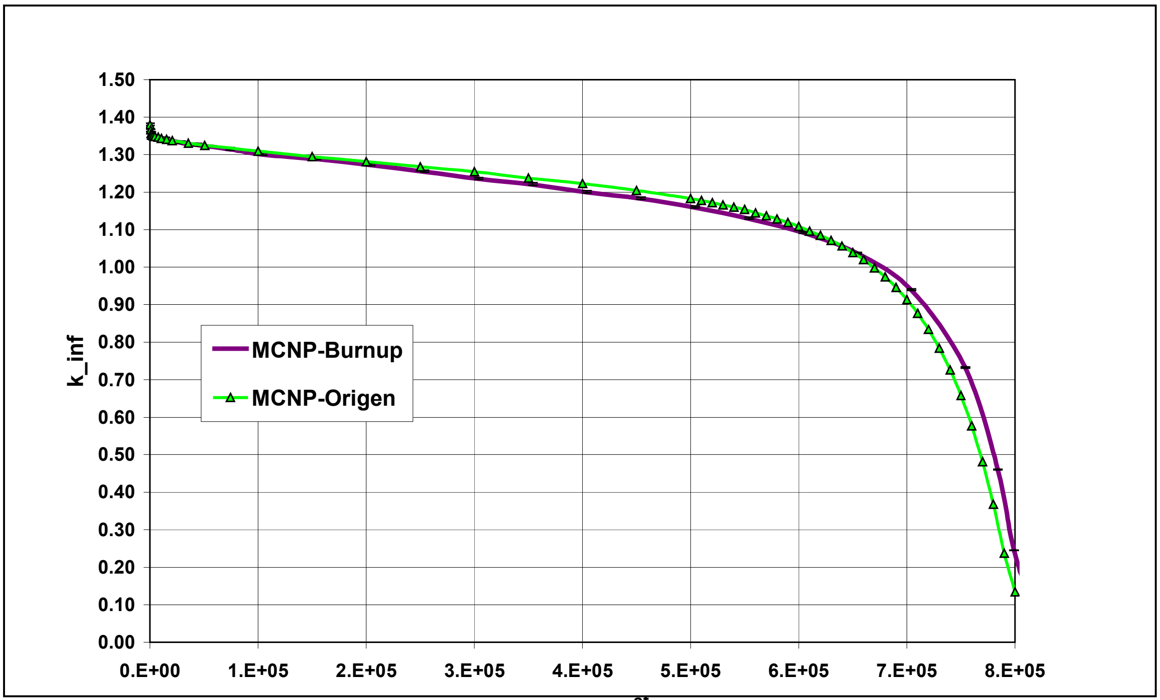

Fig. 6: k-inf as a function of burn-up for cell with 1st generation Pu, 1g Pu/pebble, Pu composition A

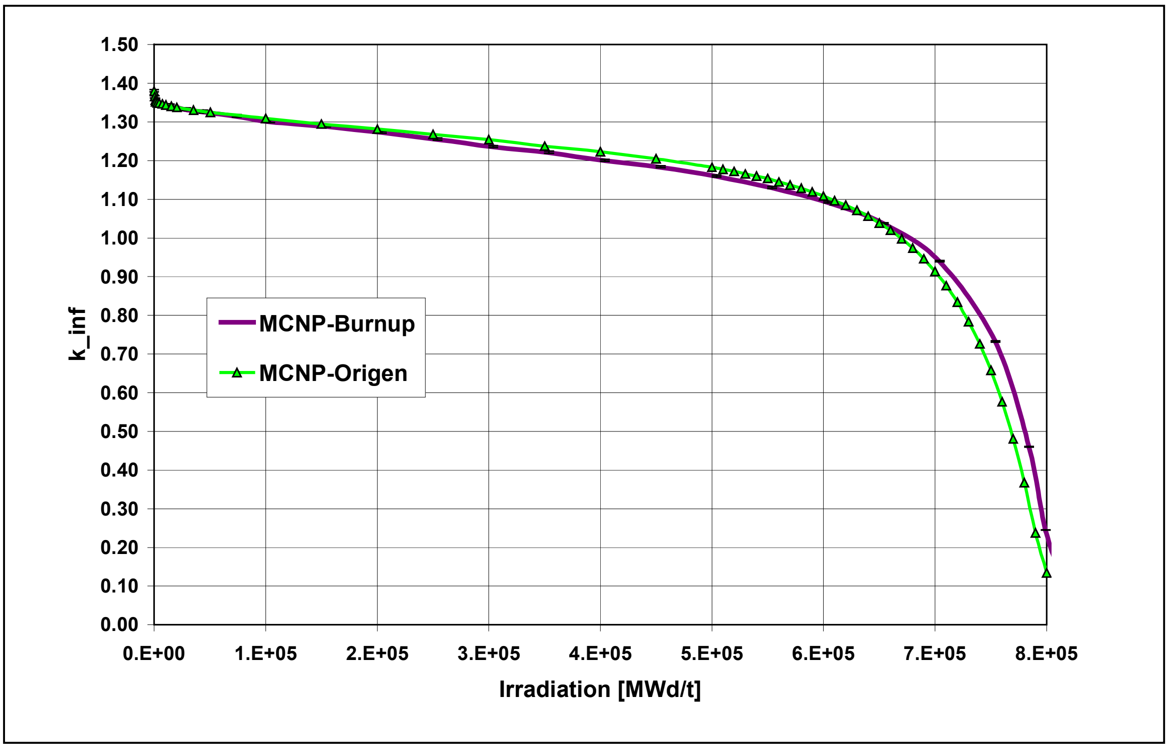

Fig. 7: k-inf as a function of burn-up for cell with 1st generation Pu, 1g Pu/pebble, Pu composition B

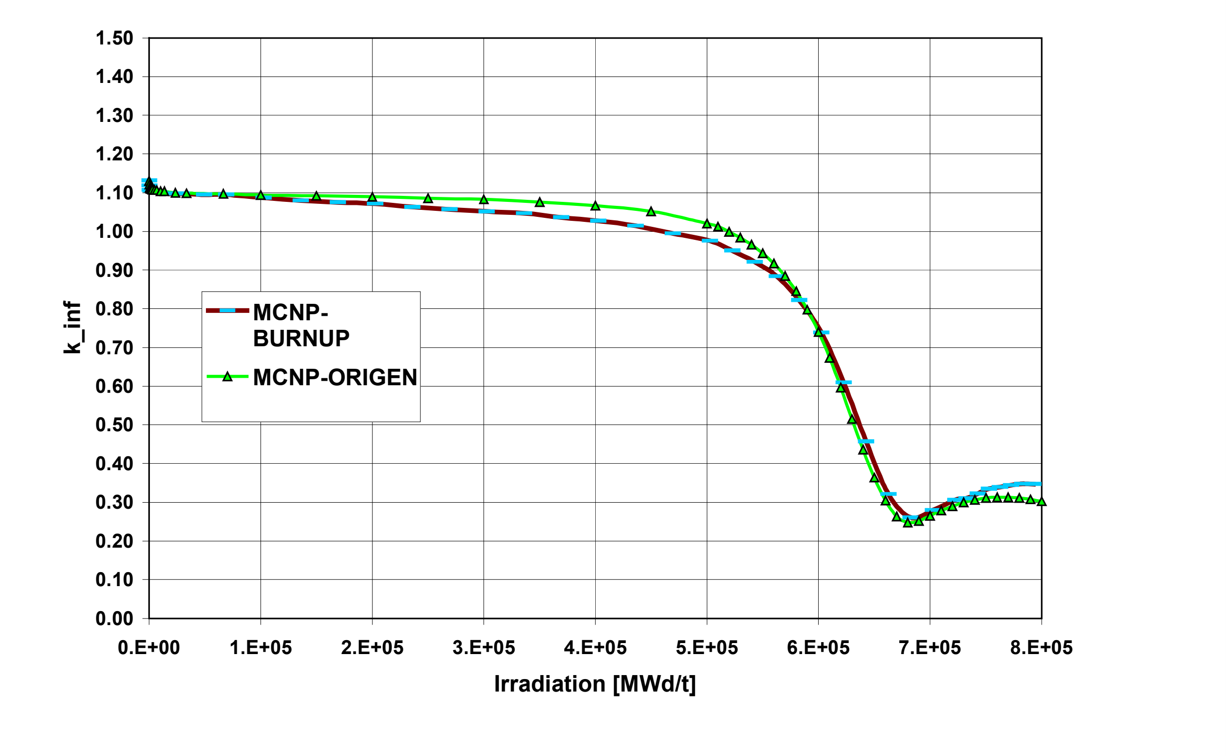

Fig. 8: k-inf as a function of burn-up for cell with 2nd generation Pu, 1,5g Pu/pebble

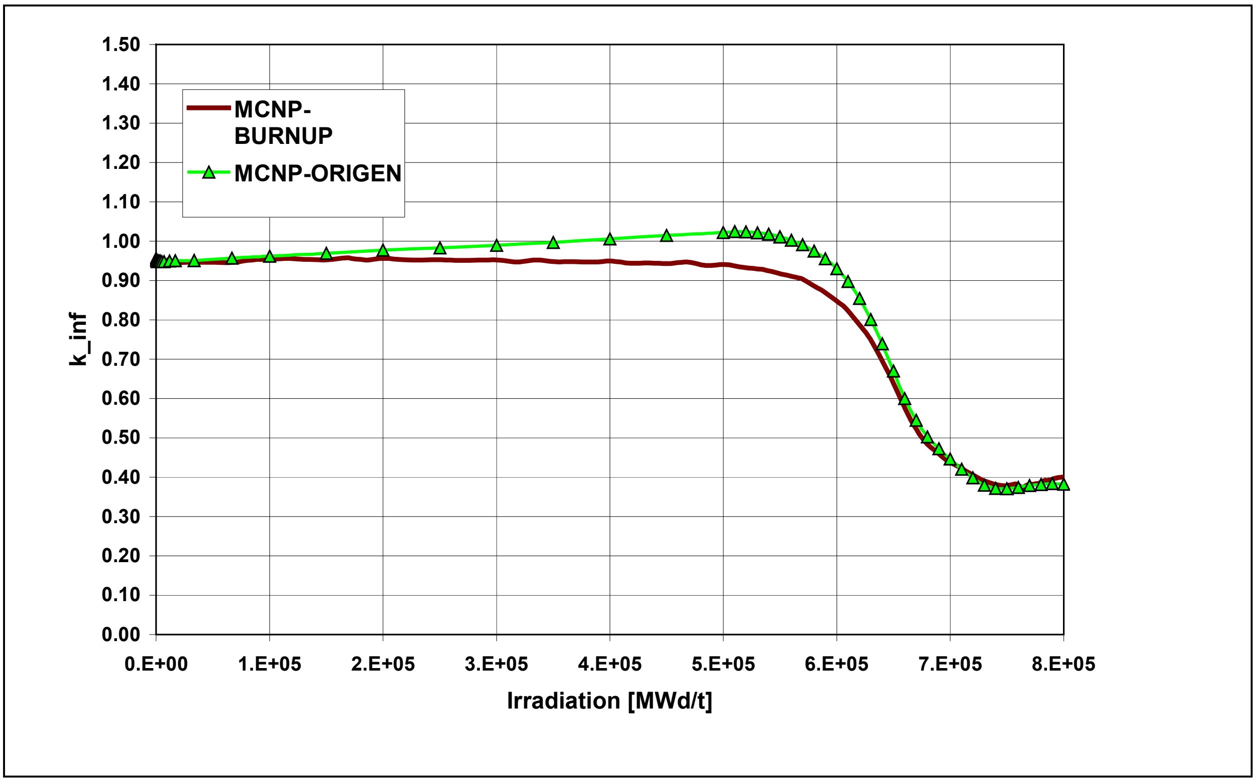

Fig. 9: k-inf as a function of burn-up for cell with 2nd generation Pu, 3g Pu/pebble

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