EVALUATION OF EFFICIENCY OF AXIAL PROFILING IN WWER-440 FUEL ASSEMBLIES

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ABSTRACT

This paper deals with application of axial profiling of fuel enrichment in WWER-440 fuel assemblies. Improvement of efficiency in fuel utilization is studied on an example of implementation of axial profiling in the second-generation fuel assemblies (FA).

To simulate fuel loads the code package SAPFIR_95&RC is used permitting to consider correctly the features of change in FA design. The methodological approach to evaluating efficiency of implementation of axial profiling using the capabilities of the above package is under consideration.

Recommendations for application of axial profiling of fuel enrichment in WWER-440 fuel assemblies are given in the conclusion.

1 INTRODUCTION

Gaining the WWER-440 operational experience promoted the step-by-step activities in improvement of indices on nuclear fuel utilization. Such measures as increase in the number of years of FA irradiation in the core, decrease in radial leakage due to fuel management scheme "in-in-in-out" have received a wide application in the reactors. Only transition for the reduced radial leakage improves efficiency of fuel utilization by 4% /1/. Further improvement of efficiency assumes decrease in axial neutron leakage due to implementation of the axial blankets of low-enriched or depleted uranium.

In evaluation of efficiency in implementation of axial blankets into the WWER-440 fuel assemblies it is necessary to take account for the design features of the structure under consideration. In particular, gas compensating plenum and water gap in the space between the FA cap and the fuel bundle upper boundary lead to generation of neutron field with "softer" spectrum in the blanket location. Use of calculation model intended for correct consideration of similar effects shall define a degree of their influence upon efficiency of fuel utilization expressed, for example, in effective full power days (EFPD) for the considered fuel load.

2 DESCRIPTION OF PROCEDURE AND INITIAL DATA

To solve the stated problem the code package SAPFIR_95&RC /2/ developed in FSUE NITI named after A.P.Alexandrov was used. The design features of the computer code SAPFIR_95 for preparation of small-group constants are usage of the generalized subgroup approach for description of the neutron energy resonant area and application of the universal geometrical modules to solve the transport equation by the method of first-collision probability (FCP). The code RC is intended for 3D calculation of the neutron field and reactor power in small-group diffusion approximation.

Let's evaluate efficiency from application of axial profiling on the example of equilibrium fuel cycle of WWER-440 with the second-generation FA to be implemented at Kola NPP, and at NPP in Czechia and Slovakia. Table 1 gives brief description of the FA types used.

Table 1 Characteristics of FA

FA type	Average fuel enrichment, %	Burnable absorber	Pitch between fuel rods, cm	Thickness of jacket walls, mm
WA	4,25	Gd_2O_3	1,23	1,5
ERC FA	3,82	-	1,23	1,5

WA – working assembly
ERC FA – emergency, regulating, compensating fuel assembly

Figure 1 shows the scheme of dividing the WA cell for calculation by the FCP method. The small-group constants for fuel-free cells are prepared with account for their surrounding by the core fragment. Figure 2 shows the scheme of dividing the ERC FA transition section at an elevation of gas compensating plenum of the fuel rod. Capabilities of the code SAPFIR_95 permit to account for the change in Hf concentration in absorbing plates on the internal surface of ERC FA jacket when preparing the small-group constants depending on irradiation time, and also to perform 3D calculations of fuel assemblies.

Figure 3 presents Hf-177 isotope concentration and relative rate of absorption reaction (SA) in the plate versus fuel burnup in the WA surrounding the ERC FA transition section. Data in the figure can be interpreted as follows: with specific power of the WA layer near the transition section equal to 84,6 kW/l and being averaged throughout fuel lifetime the Hf absorbing ability will be decreased twice in 1600 EFPD.

Two versions are considered in preparation of small-group constants of the FA blanket part using the computer code SAPFIR_95. The first version (v1) assumes 3D calculation with detailed description of fuel rod layers with fuel of main enrichment, blanket and FA fragment above the fuel bundle. In the second version (v2) the small-group constants are prepared in the "standard" 2D problem with account for leakage as the buckling approximation. The model of 3D calculation is shown in Figure 4 wherein axial distribution of fast neutrons (f1) and thermal neutrons (f2) in the upper part of blanketed WA is presented.

Generation of "softer" spectrum of neutrons in the blanket exerts influence upon time history of fission isotopes Pu (Figure 5), that in turn has influence upon $K_{infinity}$ (Figure 6). It is logical to assume that difference in $K_{infinity}$ for version v1 and version v2 becomes less with increase of blanket height and its fuel enrichment.

To define a degree of influence of "spectral" effect the comparative calculations of the blanketed reactor were performed. First, their small-group characteristics were prepared according to the first version of calculation (v1), then – according to the second version (v2). In both cases, the blanket with 0,7 % fuel enrichment in U-235 was considered, axial profiling was performed only for the WA, the blanket height amounted to 10 cm. Thus the estimated difference in duration of equilibrium fuel loads appeared to be very insignificant - less than 1(one) EFPD. Further, when preparation of the constants according to version v1 seemed to be time-consuming, the 2D calculation was performed.

To evaluate efficiency of implementation of axial blankets the comparative calculations of fuel loads are performed with the unchanged fuel management scheme, boundary conditions and parameters intended to control solution of the reactor problem. When axial profiling is performed with the fuel column total height unchanged, the average enrichment of makeup FA is kept due to proportional enrichment increase in the fuel column mid-part. Thus, annual consumption of natural uranium (G, g) is fixed at a level required to fabricate fuel for this load.

3 RESULTS

Annually, 72 fuel assemblies are loaded in the "initial" equilibrium fuel cycle, of them 60 WAs and 12 ERC FAs are loaded every odd year, and 66 WAs and 6 ERC FAs are loaded every even year. Calculation is performed with the constant values of power and coolant temperature at the core inlet when the working group of ERC FAs is at the height of 85 % from the core bottom. Average calculated duration of equilibrium load was 315,7 EFPD. Figures 7 and 8 present some characteristics of fuel load.

To determine the increment value in duration of load (dT) versus initial fuel enrichment in blankets the calculations were performed at fixed height of the blanket (Figure 9) with fuel enrichment in U-235 from 0,3% to 3,6%. These calculations have shown very insignificant dependence of value of effect upon enrichment (Figure 10). The least increment dT was obtained with blanket enrichment equal to 3,6 %.

With fixed enrichment in blanket fuel the calculations were performed for determination of dT versus height of the blanket. When height of the blanket with 0,7 % fuel enrichment in U-235 increases from 10 to 20 cm, the value of effect (dT) was increased more than twice

(Figure 11). Further increase in the blanket height (with average fuel enrichment in FA unchanged) results in necessity to increase 5% enrichment in U-235 in the central part of fuel column.

Under consideration is possible replacement of the steel rods in the ERC FA fuel rods by the blanket fuel with enrichment in U-235 up to 1,6 %. The preliminary calculation analysis /1/ shows that such replacement does not result in additional burst of power on the fuel upper boundary in the ERC FA fuel rods included into the reactor control group. When the steel rods (Figure 9) is replaced by the blanket fuel with 1,6% enrichment, the average calculated duration of equilibrium fuel loads is 322,9 EFPD. With this specific consumption of natural uranium ($G_{spec.}$, g /MW*day) was decreased by 2% with respect to “initial” fuel load.

Using the above procedure, efficiency of implementation of axial blankets was evaluated for three-year equilibrium fuel cycle. With this, enrichment in U-235 of the WA loaded annually was 3,6%, enrichment in the ERC FA - 2,4%. In fuel cartogram under consideration the first-year irradiated FAs were located over the core periphery. Effect dT from implementation of axial blankets into this fuel cycle appeared to be more significant as regards the previous fuel cycle with reduced radial leakage and refuelling ratio 4,85. For the blanket with height 10 cm and 0,7 % initial enrichment in U-235, the value of effect dT was 3 %.

4 CONCLUSION

This paper deals with study of improvement in efficiency of WWER-440 fuel utilization by implementation of axial blankets with low-enriched fuel to reduce neutrons leakage from the core. The following results were obtained using the code package SAPFIR_95&RC:

- when the blanket height is fixed the increment effect in duration of equilibrium load (dT) with unchanged annual consumption of natural uranium (G, g) practically does not depend on blanket enrichment in a range of its change from 0,3 up to 2,4% in U-235;

- with blanket enrichment being fixed and with unchanged annual consumption of natural uranium (G, g) the blanket height essentially influences the value of effect (dT);

- relative advantage of fuel efficiency with implementation of axial blankets is determined to a great extent by fuel cartogram; the fuel loads being optimized as regards reduction of radial neutron leakage are less “sensitive” to implementation of axial profiling;

- application of axial blankets in WWER-440 fuel assemblies with fixed annual consumption of natural uranium increases duration of equilibrium fuel load to 2-3%.

REFERENCES

1 Yu.A.Ananjev, G.L.Ponomarenko et al. *"Axial blankets and burnable absorber concentration profiling in WWER-440 fuel assemblies "*, The 2nd Scientific and Technical Conference "Safety Assurance of NPP with WWER", OKB GP, 2003.

2 V.G.Artemov, A.V.Elshin, A.S.Ivanov et al. *" Development of neutron-physical models of various reactor types on the basis of SAPFIR unified algorithms"* Materials of the 10th International Seminar on Reactor Physics Problems, Moscow, September 2-6, 1997, p. 34.

APPENDIX

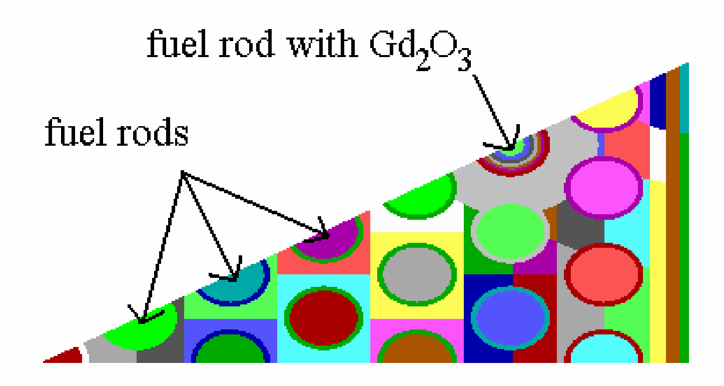

Figure 1: Scheme of dividing the WA cell for calculation by the FCP method

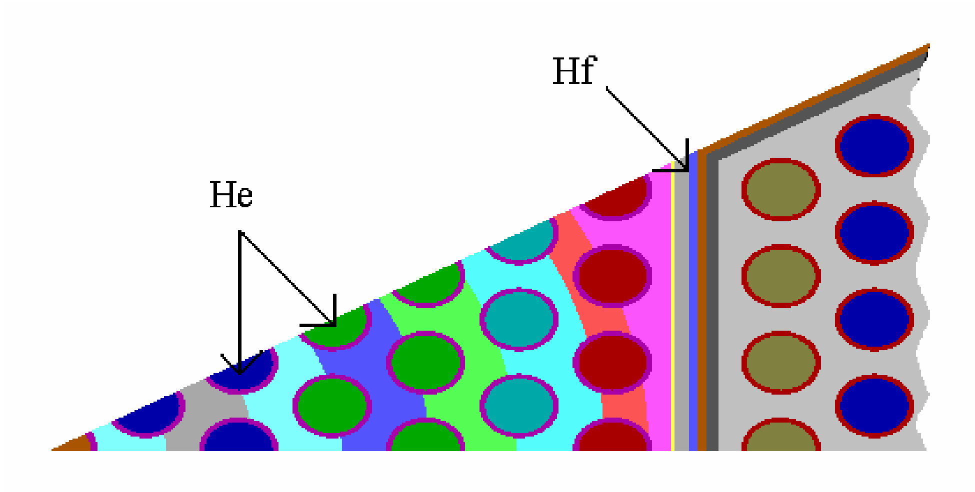

Figure 2: Scheme of dividing the ERC FA transition section cell for calculation by the FCP method

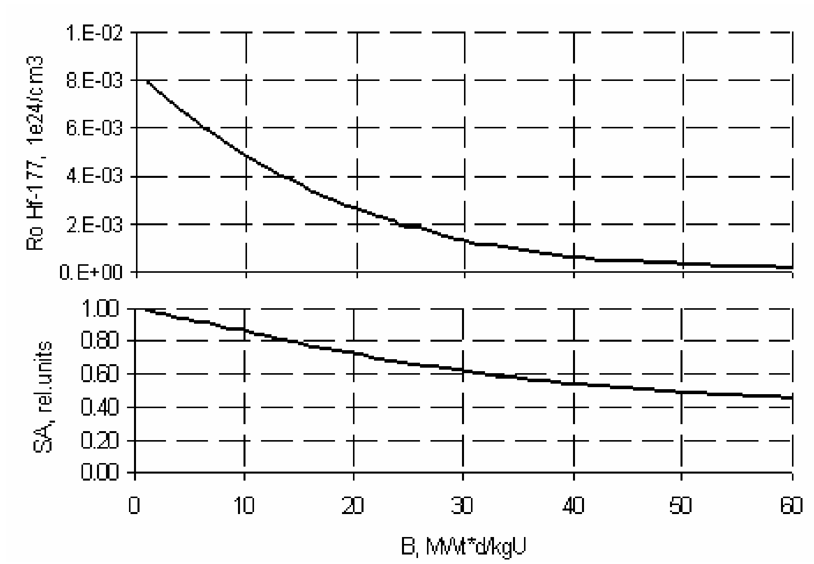

Figure 3: Change in properties of the ERC FA absorbing plate during burnup of surrounding WA

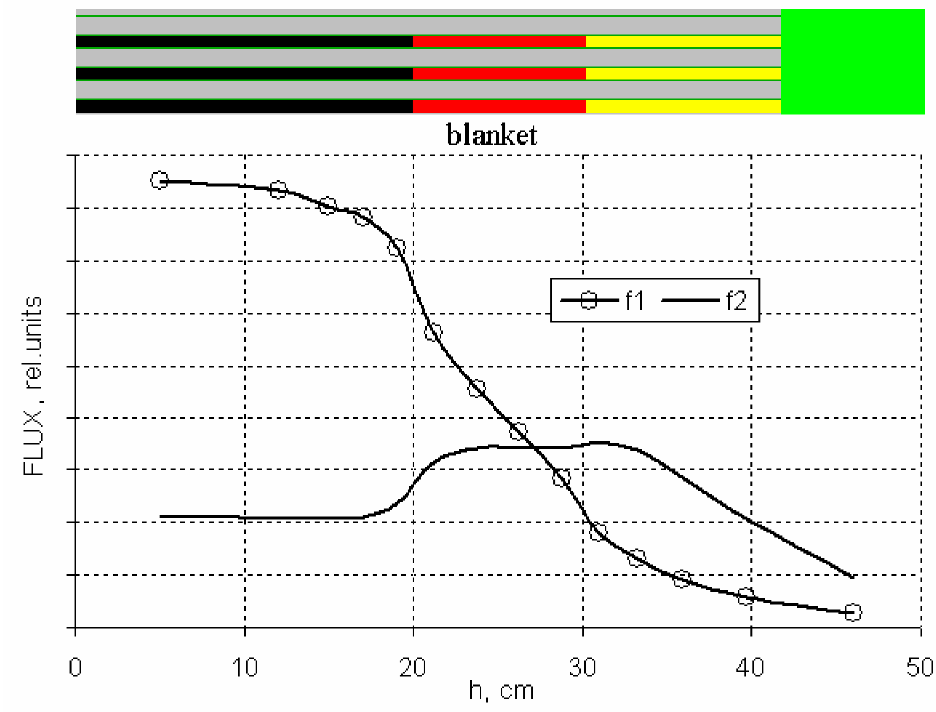

Figure 4: Three-dimensional calculation of small-group constants; axial distribution of fast neutrons (f1) and thermal neutrons (f2) in the upper part of blanketed WA

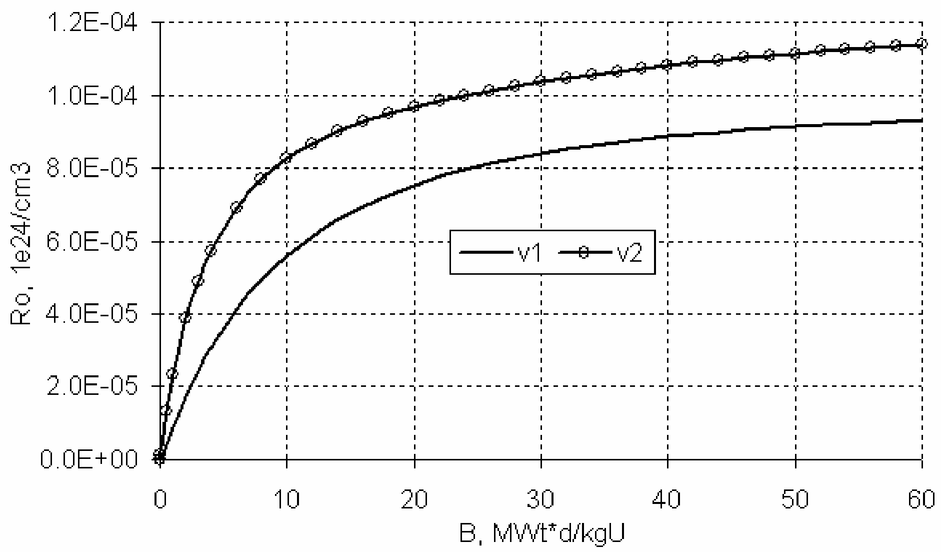

Figure 5: Change in isotope Pu-239 concentration for three-dimensional (v1) and two-dimensional (v2) versions of calculation of the blanket constants

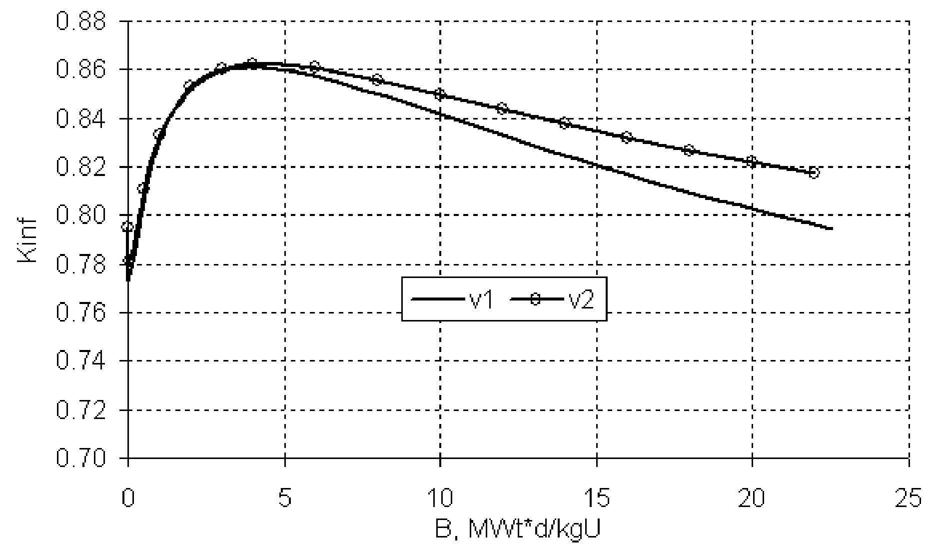

Figure 6: Change in $K_{infinity}$ of blanket with initial enrichment 0,7 % in U-235 for three-dimensional (v1) and two-dimensional (v2) versions of calculation

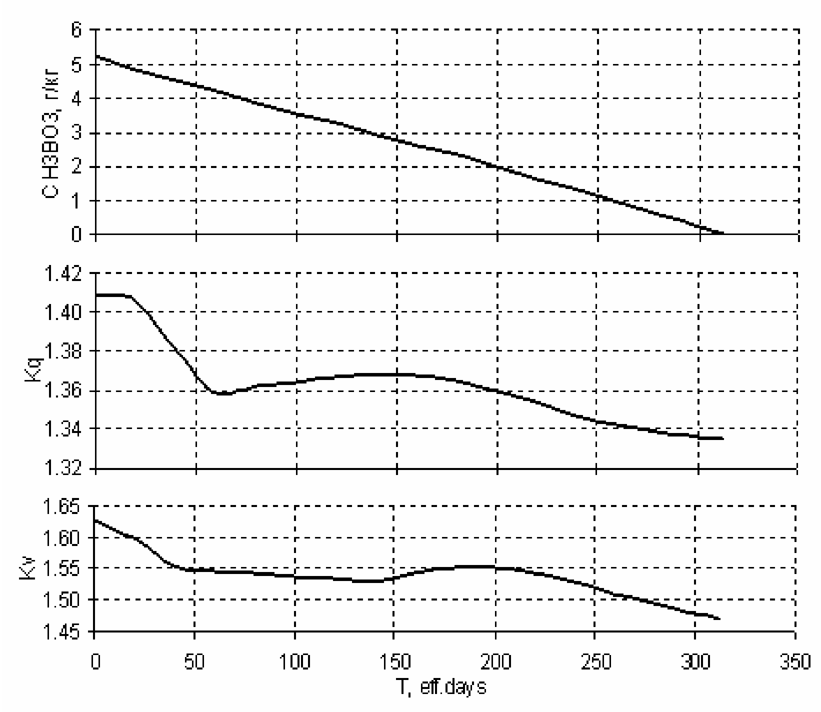

Figure 7: Change in core characteristics during equilibrium load

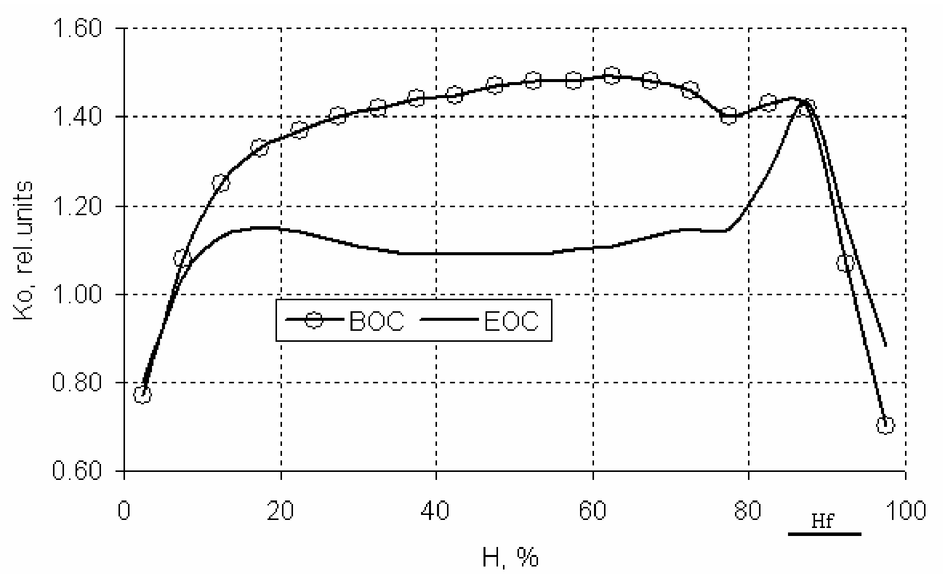

Figure 8: Axial distribution of relative linear heat rate of peripheral fuel rod of WA near ERC FA working group at BOC and EOC

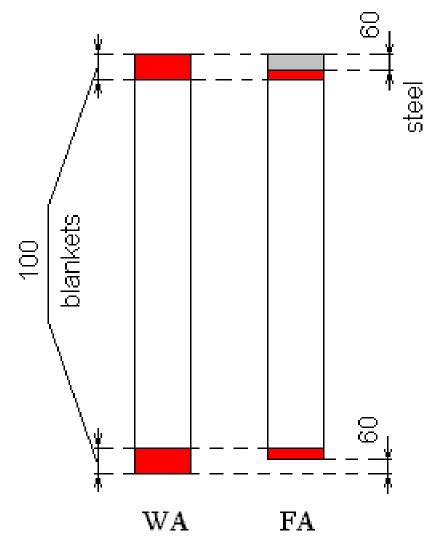

Figure 9: Axial fuel profiling in fuel rods of WWER–400 fuel assemblies

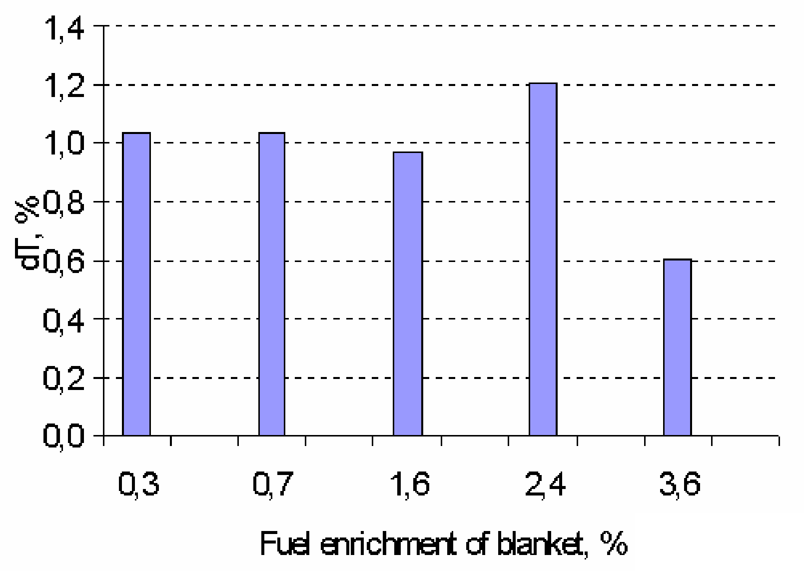

Figure 10: Increment effect in duration of equilibrium load (dT) of five-year fuel cycle with average enrichment in makeup fuel versus blanket enrichment; blanket height is 10 cm

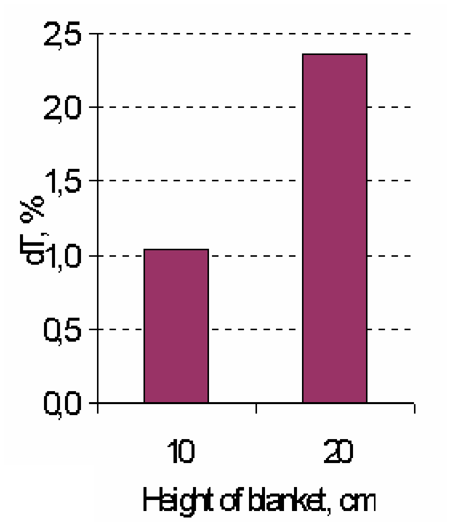

Figure 11: Value of increment effect in duration of equilibrium load (dT) of five-year fuel cycle with average enrichment in makeup fuel versus blanket height; blanket enrichment in U-235 is 0,7 %