Neutronic Calculations Regarding the New LEU 6x6 Fuel Bundle for 14 MW TRIGA –SSR, in Order to Increase the Reactor Power up to 21 MW

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ABSTRACT

In order to meet the increasing demands of terminal flux for the experimental devices which will be loaded with CANDU natural uranium pins (or clusters), is necessary to rise the reactor power up to 21 MW.

In this respect we consider in our evaluations a new 6x6 TRIGA fuel bundle geometry (the actual fuel bundle contains 5x5 pins)

The paper will contain a comparative analysis regarding: flux and power distribution across the 29 fuel bundles standard core, and fuel managements patterns, in order to maximize the discharge fuel burnup and core lifetime.

INTRODUCTION

The Romanian 14 MW TRIGA research and test reactor was commissioned in 1979 by General Atomics U.S.A. It was specifically designed to test in-core fuel assemblies and loops. The core was designed to produce high neutron flux in the several experimental locations with experiments in place.

The standard core configuration (Figure 1) using 5x5 fuel bundles (Figure 2a) has a multipurpose nature, this being indicated by the following features:

- six in-core experimental locations [3 large (17.5x17.5 cm^2) and three small (8.75x8.75 cm^2)]
- two horizontal beamports (tangential, radial)
- numerous irradiation facilities in berilium reflector blocks

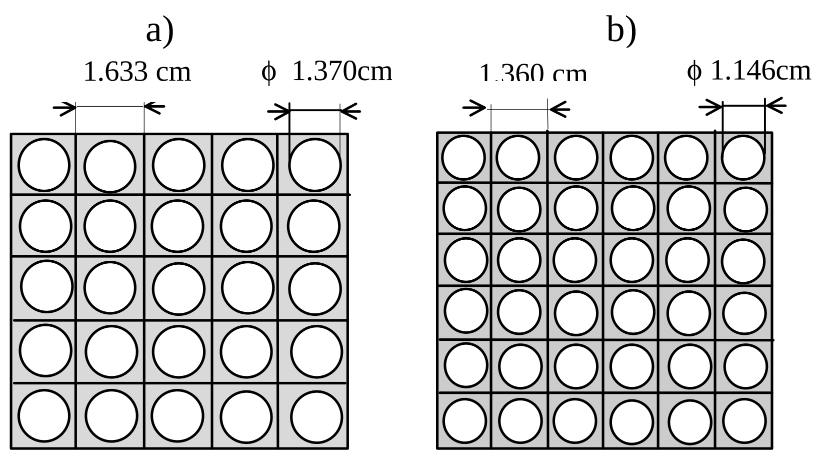
Usually there are two or three in-core devices containing experiments with low enriched (2%-5%) CANDU fuel type rods. Although the core configuration is not optimum for the production of leakage fluxes, the reactor has a radial beamport with a neutron spectrometer attached.

In order to meet the increasing demands of experimental specifications which often need a higher thermal flux than we can usually get, we consider the alternative of rise the reactor power up to 21 MW using a new fuel bundle concept containing 6x6 LEU fuel pins. (Figure 2b)

This concept was enhanced by General Atomics [1] since 1992. This paper is intended to provide analytical support and to develop the concept in order to meet new experimental demands.

FIGURE 2

5X5 & 6X6 Fuel Bundle Layout



TECHNICAL FEATURES OF 6x6 FUEL BUNDLE

The excellent performance and safety features of the TRIGA fuel during the 19 years of operation and the high burnup reached by HEU & LEU encouraged us to develop this concept of fuel that can bear higher power densities on fuel bundle.

In the present the cooling system of the reactor has four pumps and three heat exchangers, 7 MW each, there is room to add one supplementary heat exchanger. Together three pumps and three heat exchangers can remove 21 MW thermal power.

The calculations carried out in this paper evolve from some conservative considerations as follows:

- the coolant flow cross section in 6x6 fuel bundle has the same value like 5x5 fuel bundle in order to preserve the initial designed inlet flowing conditions

- the fuel clad for 6x6 concept is made of Incoloy 800, the same material like 5x5 case

- the fuel length and gap between fuel and clad was also preserved at the initial value in order to maintain the initial conditions

- thermal conductivity properties were considered the same because we use the LEU fuel type

Using the above conservative conditions one can extract the pin dimensions The comparison between 5x5 and 6x6 fuel bundle is shown in TABLE 1

TABLE 1

Core/Fuel Bundle	LEU 5x5	LEU 6x6
Center to center distance	1.633 cm	1.360 cm
Fuel radius	0.647 cm	0.573 cm
Outer clad radius	0.688 cm	0.613 cm
Gap	2.22 E-03 cm	2.22 E-03 cm
Clad thickness	0.04 cm	0.04 cm
Total # fuel bundles	29	29
Total # in-core pins	725	1044
Coolant flow rate	4.73 E+2 l/s 2 pumps	6.88 E+2 l/s 3 pumps
Total mesh area	2.666 cm^2	1.849 cm^2
Fuel area	1.317 cm^2	0.889 cm^2
Clad area	0.172 cm^2	0.140 cm^2
Water area	1.176 cm^2	0.960 cm^2

CALCULATION OF MICROSCOPIC X-SECTIONS

In order to calculate the X-sections which are necessary in the burnup loop, is used a modified version of WIMS-D code which calculate the microscopic X-sections for 54 nuclides and one fission pseudoproduct. The list of those nuclides is shown in TABLE 2.

TABLE 2

U234	U235	U236	U238	PU239	PU240	PU241	PU242	KR83
MO95	TC99	RU101	RU103	RH103	RH105	PD105	PD108	AG109
CD113	IN115	I127	XE131	CS133	CS134	XE135	CS135	ND143
ND145	PM147	SM147	PM148m	PM148	SM149	SM150	SM151	SM152
EU153	EU154	EU155	GD157	FISP	ER166	ER167	H(HZR)	H(H2O)
OXY	AL	ZR	FE	CR	NI	MN	SI	C
B10								

The cell calculations are based on the average pin data. The initial LEU fuel pin loadings Wt% and atom densities are illustrated in TABLE 3

TABLE 3

Element/Isotope	LEU 6x6 (Wt%)	LEU 6x6 (atoms/cm-barn)
U	45.0	9.3723 E-03
234	0.15	1.4263 E-05
235	19.7	1.8652 E-03
236	0.25	2.3569 E-05
238	79.9	7.4693 E-03
Er	1.074	2.6236 E-04
166	33.33	1.0665 E-04
167	22.90	7.2811 E-05

The transport calculations has been made using WIMS code on 23 broad groups and than collapsed on 7 broad groups which are used in tridimensional diffusion calculations. Infinite multiplication factor for the (5x5) fuel bundle was found : **K-inf.=1.3751** and for the (6x6) fuel bundle it is : **K-inf.=1.3837.**

As one can notice the infinite multiplication factor (K-inf.) is slight greater for the 6x6 case. The cell calculations covered the burnup range between 0-20000 MWD equivalent core.

In the above mentioned range seven sets of microscopic X-sections have been obtained at 20 °C and another seven sets at 230 °C for LEU 6x6 fuel, which are used for the core burnup calculations.

CORE CALCULATIONS

For the core configuration which will use the new 6x6 LEU fuel bundle, some comparisons with the core using the old 5x5 fuel bundle have to be made. In order to cover this task, some neutronic and thermohydraulic calculations should be done. For the neutronic calculations we use a tridimensional diffusion code (**DFT**) with burnup loop. DFT was developed and tested at Institute For Nuclear Research . The DFT code uses an identical burnup scheme like WIMS does, that because the burnup pseudoproducts have to be compatible each-other for the cell and core calculations.

The burnup calculations cover the range between 0-14000 MWD for the 15 MW thermal power. Thermohydraulic evaluation has been performed using **PARET** code (hydrodynamic, point cinetic, code provided by A.N.L. U.S.A.). Using the highest pin factors, from neutronic evaluation for 6x6 fuel bundle core, APF and PPF were obtained :

APF (Axial Peaking Factor) = 1.30
PPF (Power Peaking Factor) = 2.02

The maximum temperatures for the hotest spot from the core are shown for some power levels in TABLE 4

TABLE 4

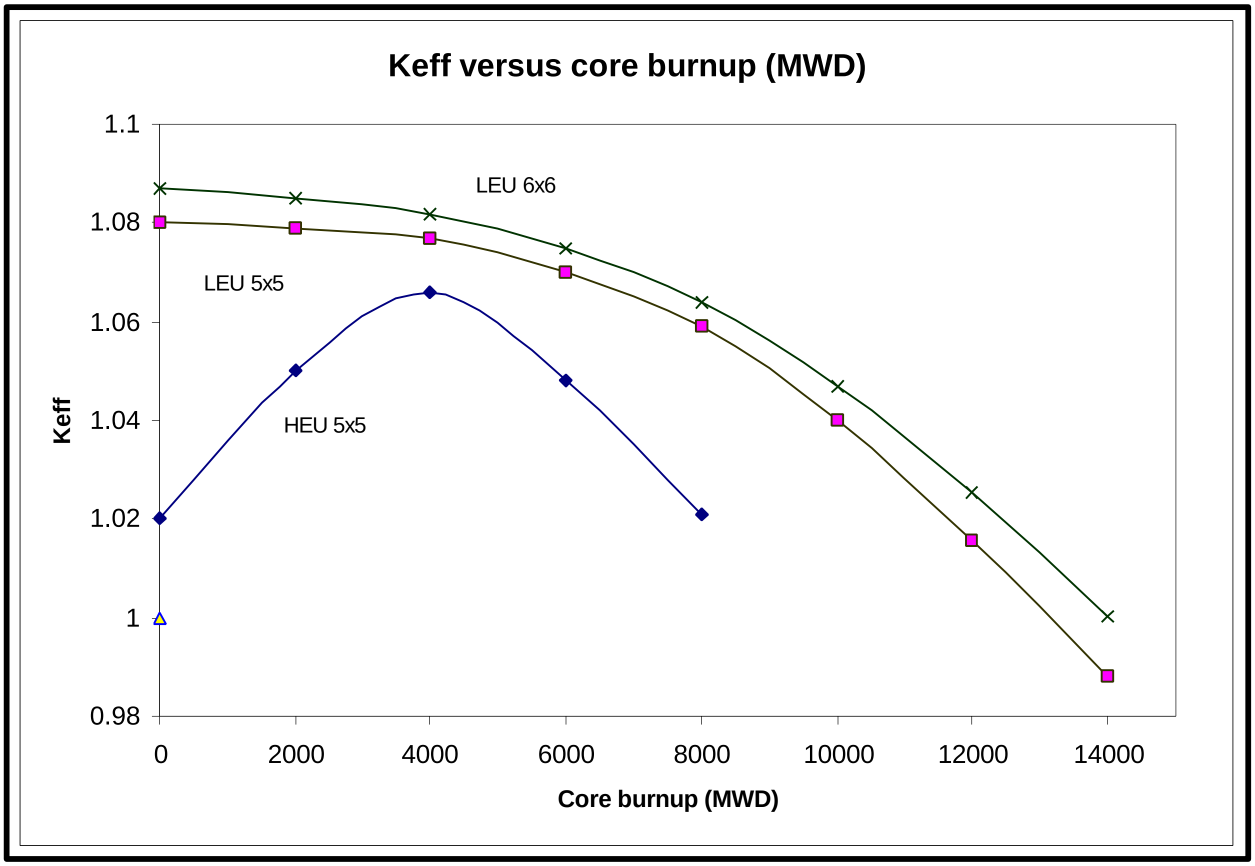
P-reactor (MW)	T C (2 Pumps)	T C (3 Pumps)
10	389	-----
14	529	-----
21	738	726
25	911	896

The main results of core calculations are contained in TABLE 5, and the K_{eff} evolution versus burnup in FIGURE 3

TABLE 5

Parameter	Core with 5x5 HEU	Core with 5x5 LEU	Core with 6x6 LEU
K_{eff} max (at 20 °C unpoisoned)	1.104	1.115	1.119
K_{eff} max (at 300 °C poisoned at 15 MW)	1.065	1.08	1.087
APF max.	1.3	1.3	1.3
PPF max.	2.3	2.03	2.02
Maximum center pin temperature °C	600 (at 15 MW)	550 (at 15 MW)	726 (at 21 MW) 829 (at 25 MW)
Core lifetime (MWD)	8500	12000	13500
Maximum thermal flux < 1.125 eV	2.9 E+14 nv (at 14 MW)	2.75 E+14 nv (at 14 MW)	4.30 E+14 nv (at 21 MW)

FIGURE 3



As one can see from above data presentation the core using 6x6 LEU fuel bundle can be operated safely at 21 MW (from maximum temperatures point of view, the safety temperature limit for normal operation is set at 750 °C by the fuel provider G.A.)

The K_{eff} evolution versus core burnup presented in Figure 3 shows that the core life time is longer in 6x6 case (14000 MWD till first fuel shuffling) compared to HEU 5x5 (8500 MWD till first fuel shuffling) and LEU 5x5 (12500 MWD till first fuel shuffling) case. The K_{eff} low limit has been taken at 1$ reactivity (β =0.007) , the fuel 6x6 is used in a more efficient manner.

According to above mentioned data, it follows that among the solutions presented the fuel bundle containing 6x6 pins is far more advantageous. Although in the new designed core are much more pins (1044 compared with 725) the PPF is not much less (2.02 compared with 2.03) that's because the strong self shadowing effect of the inner pins in the 6x6 fuel bundle.

The calculated control rod bank worth (the reactivity between bank fully withdrawn and fully inserted is aprox 19$) [2] [3] is sufficient to cover the maximum reactivity excess (K_{eff} max =1.119 aprox. 16.8$)

CONCLUSION

This new concept provides higher flux for in-core experiments as well as for beamports and in the same time the expected burnup is greater using the same initial loadings for fuel pin and the same burnable poison. These facts lead to certain economic advantages.

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