Burnup Analysis of Fuel Assembly Designs for the 300 kW Small Medical Reactor

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1. Introduction

A 300 kW small medical reactor was designed to be used for boron neutron capture therapy (BNCT) at KAIST in 1996 [1].

In this paper, in order to extend a core life cycle modifications of fuel assembly design with high density fuel for the BNCT facility were performed and a criticality, neutron flux distribution and fuel burnup calculations were carried out.

2. Design Modifications

The reactor core is composed with four slab assemblies and D_2O tank in center and this NCT facility provides four neutron beam tubes. Thermal and epithermal neutron beam ports are provided separately. The quarter midplane horizontal section of the proposed facility is shown in fig 1(Model 1). There have been designed two models of the reactor core. The difference between these two models is in the air part of each irradiation port. Each fuel assembly has 7x79 hexagonal lattice and the whole reactor core consists of 1056 fuel rods and initially 36 control rods.

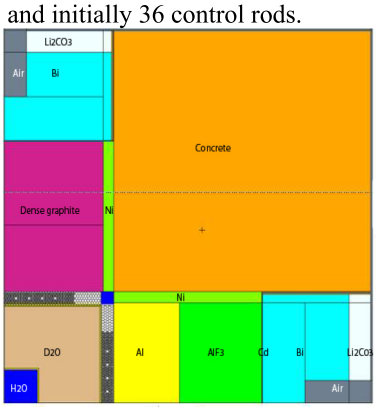

Fig 1. Quarter midplane horizontal section of the proposed facility

A variety of alloy type high densification uranium fuels such as U-7Mo and U-4Zr-2Nb which have very low neutron absorption cross section are under development [2, 3]. In this work the high density fuels were used to lengthen the core life cycle at the same 300 kW power level and to get high quality and high flux neutron flux at particular irradiation ports. Low enrichment uranium (<20wt % 235U) fuels were used.

It needs to change the assembly configuration to maintain the critical core and more control rods should be added and high enrichment B^{10} in the control rods were used. There were carried out three test calculations; in the first one the high density fuels are directly substituted in the same fuel assembly as low density fuel used in reference [1] (T0), in the second case 17 control rods per half assembly were added (T1) and in the last one the added number of control rods a half assembly was 10 and the Boron 10 isotope was enriched up to 70 % (T2).

3. Numerical Results

The MCNP code [4] was used to carry out the criticality and the neutron transport computations needed this study. The source strength is calculated to be $2.278125*10^{16}$ fission neutrons/sec assuming 200 MeV/fission and 2.43 fission neutrons/fission.

The effective multiplication factor calculation results are listed in Table I.

Table I. Effective multiplication factor

Fuel	Test	Control rods	Model 1	Model 2
U3Si2-Al	0	In	0.93877±0.00095	0.93834±0.00096
		Out	1.04547±0.00093	1.04641±0.00086
	1	In	0.69553±0.00057	0.69509±0.00059
		Out	1.04531±0.00059	1.05251±0.00061
	2	In	0.72644±0.00054	0.72528±0.00057
		Out	1.04668±0.00059	1.05270±0.00065
U-7Mo	0	In	1.09529±0.00062	-
		Out	1.14939±0.00062	-
	1	In	0.90953±0.00059	0.90878±0.00063
		Out	1.17275±0.00061	1.17250±0.00055
	2	In	0.94167±0.00058	0.94195±0.00062
		Out	1.17157±0.00059	1.17152±0.00060
U-4Zr-2Nb	0	In	1.09964±0.00065	-
		Out	1.15523±0.00061	-
	1	In	0.91391±0.00058	0.91444±0.00063
		Out	1.18034±0.00058	1.18024±0.00064
	2	In	0.94575±0.00062	0.94782±0.00065
		Out	1.17801±0.00066	1.18076±0.00062

The flux spectrum averaged over whole core at beginning of the life cycle in various fuel cases is shown in fig 2. From this figure, when more control rods are replaced by water rods (all control rods are out) neutron spectra become more softer than the case with few control rods so that low energy portion in T1 test is greater than that in T0 test with the same fuel.

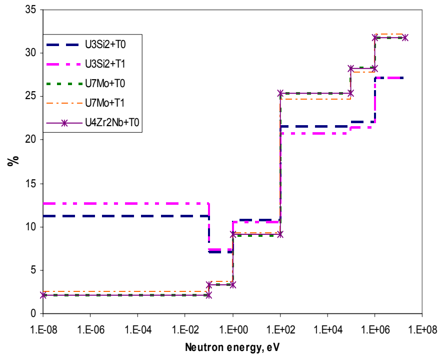

Fig 2. Flux spectrum in the beginning of the cycle in whole core

For comparative purposes, fluxes were generally tallied in three energy bins: thermal (<0.4 eV), epithermal (0.4 eV to 1 keV), and fast (>1 keV). The neutron flux distributions along the beam central line at the proposed epithermal and thermal beam tubes were shown in fig 3a and 3b, respectively.

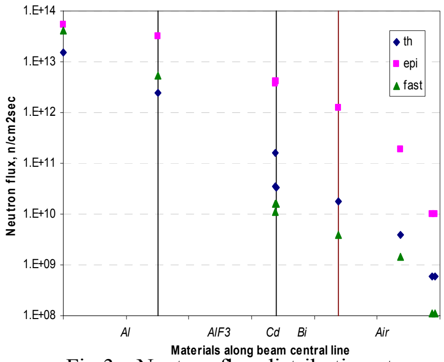

Fig 3a. Neutron flux distribution at epithermal column

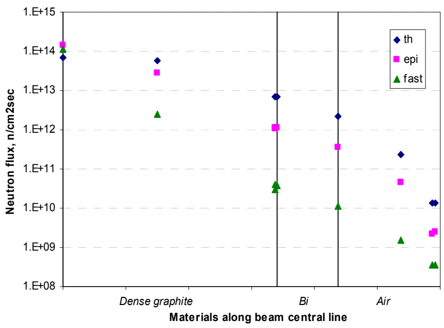

Fig 3b. Neutron flux distribution at thermal column

The calculation of the keff and its relationship with core burnup is of primary importance to determine the core lifetime. The MONTEBURNS code [5] was used to carry out the burnup and the core life cycle calculations needed in this study. The excess reactivity for the beginning of the core life and the life cycle were calculated by MONTEBURNS at 300kW reactor power and results are listed in Table II for various fuel cases. The burnup calculation was performed for the small medical reactor core without changing the loading pattern until the excess reactivity fell to zero. Figure 4 presents the fuel burnup characteristics.

Table II. Excess reactivity in the beginning of the cycle, % and acceptable life cycle, day

Fuel	Test	Model 1	Model 2	Life cycle, day
U3Si2-Al	Initial	4.35	4.44	1
	Test 1	4.33	4.99	
	Test 2	4.46	5.01	
U-7Mo	Test 1	14.73	14.71	> 60
	Test 2	14.64	14.64	
U-4Zr-2Nb	Test 1	15.28	15.27	~ 50
	Test 2	15.11	15.31	

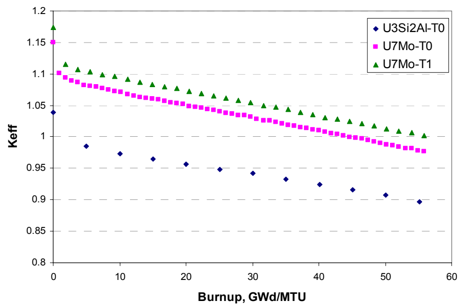

Fig 4. Multiplication factor as a function of fuel burnup

4. Conclusions

The paper provides description of modifications of fuel assembly design with high density fuel for the BNCT facility as well as a criticality, neutron flux distribution and fuel burnup calculations. The proposed facility provides an epithermal and thermal neutron beam of $9*10^9$ n_{epi}/cm^2sec and $1.3*10^{10}$ n_{epi}/cm^2sec, respectively. The core life cycle was evaluated around 50 days which is equivalent to ~ 56 GWd/MTU fuel burnup with high density fuel assembly design.

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