NEUTRONIC PARAMETERS OF TAJOURA RESEARCH REACTOR FUELED WITH *HEU* and *LEU*

Omran Abuzid Abulgasem Faisel Abdula Abuatweirat
AbduAlhamed Kh. Ajaj

Renewable Energies and
Water Desalination Research Center
P. O. Box 30878
Tripoli Libya
e-mail omran_abuzid@yahoo.com

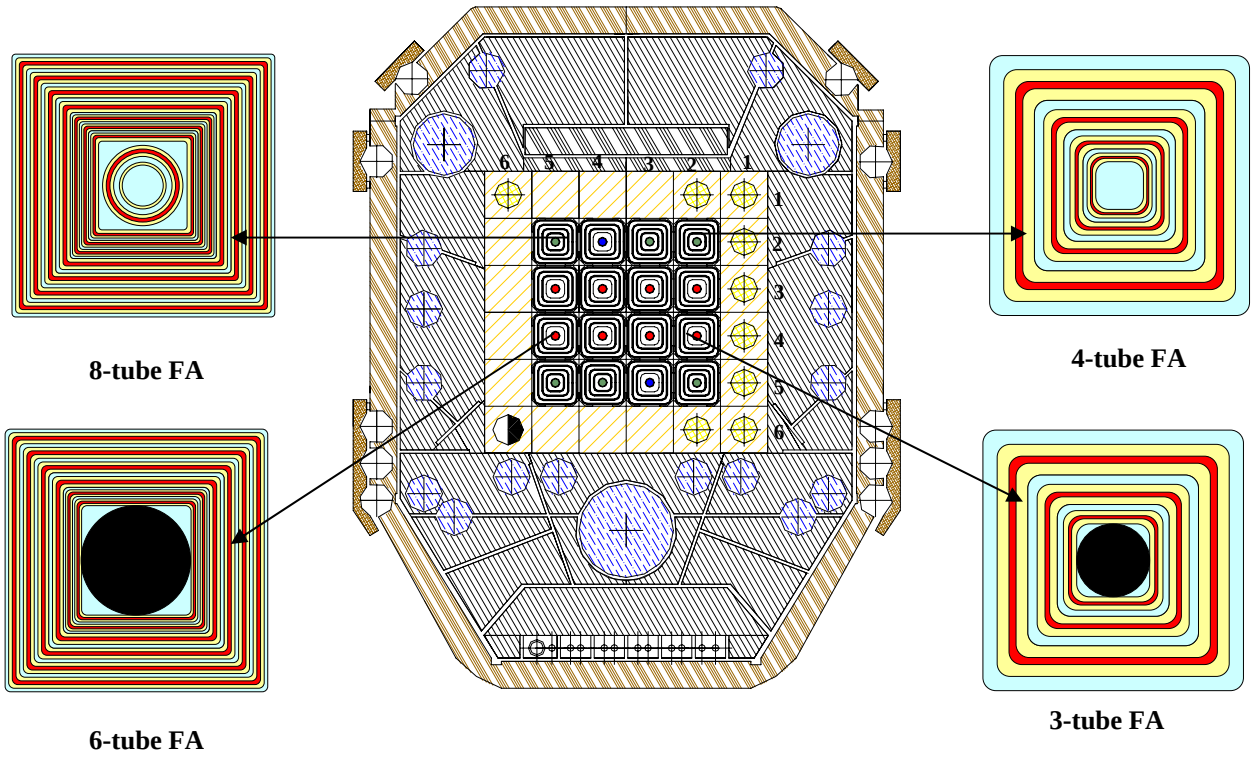
ABSTRACT

The computer code WIMSD5B is used to determine the cross sections of the reactor cells and then CITTATION diffusion computer code is used to investigate the neutronic parameters of the IRT-1 Tajoura research reactor fuelled with High Enriched Uranium (***HEU***) and Low Enriched Uranium (**LEU**). 4-tube and 3-tube of 80% enrichment, IRT-2M HEU fuel and 8-tube and 6-tube 19.7% enrichment, IRT-4M LEU fuel cells are studied. This paper presents the obtained results of the core with both fuel types.
The fresh core calculations show that fuelling Tajoura reactor core with *LEU* will increase the fast neutron flux but decreases the thermal flux (detailed values are tabulated).
The maximum values of the thermal neutron flux is increased almost three times using a central neutron trap

The core excess reactivity increases by more than 1% when the core is loaded with 16 fuel assemblies all of 6-tube type and by more than 2% when the core is loaded with 10 fuel assemblies of 6-tube and 6 fuel assemblies of 8-tube type.
The core excess reactivity for ***HEU*** is 14.02%,, for the 6-tubes ***LEU*** core it is 15.04%, and for the mixed 6 and 8-tubes core it is 16.21%.

INTRODUCTION

The 10 MW IRT-1 Tajoura reactor is a pool type reactor cooled and moderated by light water. The pitch is 7.15 cm and the active fuel length of 58 cm for the HEU fuel and 60 cm for the LEU fuel. Cross sections of 4-tube, 3-tube, 6-tube and 8-tube fuel assemblies and core layout are shown in Figure(1). The 3-tube HEU and 6-tube *LEU* fuel assemblies are to contain safety and shim control rods. The regulating rod is accommodated in a removable beryllium unit. The absorbing material in all of the rods is boron carbide enclosed in a stainless steel cladding. The stationary beryllium reflector and the aluminium vessel contain a number of vertical irradiation channels called VCR in the stationary beryllium and VCV in the aluminium vessel. The VCR channels can be plugged with beryllium plugs when they are not utilized.



Figure(1) Cross section of the 4-tube and 3-tube IR-2M 8-tube and 6-tube IR-4M fuel assemblies and core layout

CALCULATIONS

The computer code WIMSD5B with a library of 69 energy groups is used for the calculations of the different cells of the reactor. The cell calculation is carried out employing the full 69 groups and then a post processor was used to collapse the energy groups down to seven groups with energy upper boundaries of 10.0, 8.21E-1, 5.53E-3, 4.00E-6, 6.25E-7 and 2.50E-7 Mev respectively. All cell calculations were carried out using the equivalent annuli and the DSN main transport routine. In order to use the fuel spectrum for the non fuel cells calculations a homogeneous fuel region was introduced at the center of all non fuel cells and then in the post processing this extra-added region was excluded from the cell. The calculations were performed with HEU and two cases of LEU cores. Case(A) the 16 fuel assemblies are all of 6-tube type and case(B) loading 10 fuel assemblies of 6-tube and 6 fuel assemblies of 8-tube type. The values of k_{eff} and the corresponding system reactivity in $\Delta k/k$ % are given in Table(1) for. Calculations with central neutron trap were carried out with loadings of 20 fuel assemblies for the HEU core, 20, 18 and 16 fuel assemblies for the LEU core.

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Table (1) **Values of k_{eff} and $\Delta k/k$ % for *LEU* core with all control rods are fully raised**

	10 of 3 tube & 6 of 4 tube HEU FUEL		*All 6 tube LEU FUEL*		*10 of 6 tube & 6 of 8 tube LEU FUEL*	
	k_{eff}	$\Delta k/k$ %	k_{eff}	$\Delta k/k$ %	k_{eff}	$\Delta k/k$ %
all plugs are in place	1.1628780	14.02	1.1770413	15.04	1.1935079	16.21
plugs are replaced by water	1.1503427	13.07	1.1625152	13.98	1.1784456	15.14

Detailed flux mapping with HEU and LEU fuels has been obtained for the fresh core. The maximum values of fast and thermal fluxes for a full power of 10 MW are given in Table (2) for different locations in the reactor.

Table(2) Maximum values of fast and thermal fluxes

	Cell (2-2)	Cell (3-2)	Cell (4-1)	Cell (1-6)	Cell (6-1)	12VCR
HEU core $\phi_{fast_\max}$	4.0567E14	4.5740E14	1.9438E14	9.5669E13	7.5408E13	6.792E13
LEU core $\phi_{fast_\max}$	4.4065E14	5.8127E14	2.1525E14	1.0589E14	8.229E13	7.587E13
HEU core $\phi_{th_\max}$	1.7548E14	1.8761E14	3.3398E14	2.1898E14	1.9334E14	1.7212E14
LEU core $\phi_{th_\max}$	1.2449E14	1.2941E14	3.3566E14	2.2605E14	1.9983E14	1.7527E14

CONTROL RODS WORTH

the amount of positive or negative reactivity introduced into the core is usually determined from the curves of the total (integral) worth of the control rod being moved. Loading the reactor with LEU fuel requires the determination of the reactivity worth of all control elements. CITATION is used to determine the total worth of the safety, regulating and shim rods

Control rod calibration was performed with the positions of the remaining rods being very similar to experimental positions this means to start calibration with the critical. The integral worth values of the individual rods are given in Table (3). The control elements total worth for the *HEU* core is 25.9% and for case (A) of the *LEU* core is 25.2% while for case (B) of the *LEU* core it is 24.9%.

Table(3) **Control Rods total worth calculated at critical state (with all plugs in place).**

Rod	***6 FA of 4 & 10 FA of 3-tube type HEU*** core	***16 FA of 6-tube type LEU*** core	***6 FA of 8 & 10 FA of 6-tube type LEU*** core
	$\Delta k/k$ %	$\Delta k/k$ %	$\Delta k/k$ %
Safety Rod 1	2.18250	2.22990	2.21048
Safety Rod 2	2.47542	2.54397	2.54541
Regulating R	0.37567	0.39365	0.44771
Shim Rod 1	2.27039	2.30972	2.27827
Shim Rod 2	2.38291	2.40391	2.40286
Shim Rod 3	2.87654	2.67357	2.58593
Shim Rod 4	3.00973	2.77891	2.73621
Shim Rod 5	2.84704	2.61274	2.55337
Shim Rod 6	2.96888	2.76155	2.68389
Shim Rod 7	2.16553	2.16718	2.12950
Shim Rod 8	2.31922	2.33523	2.30522

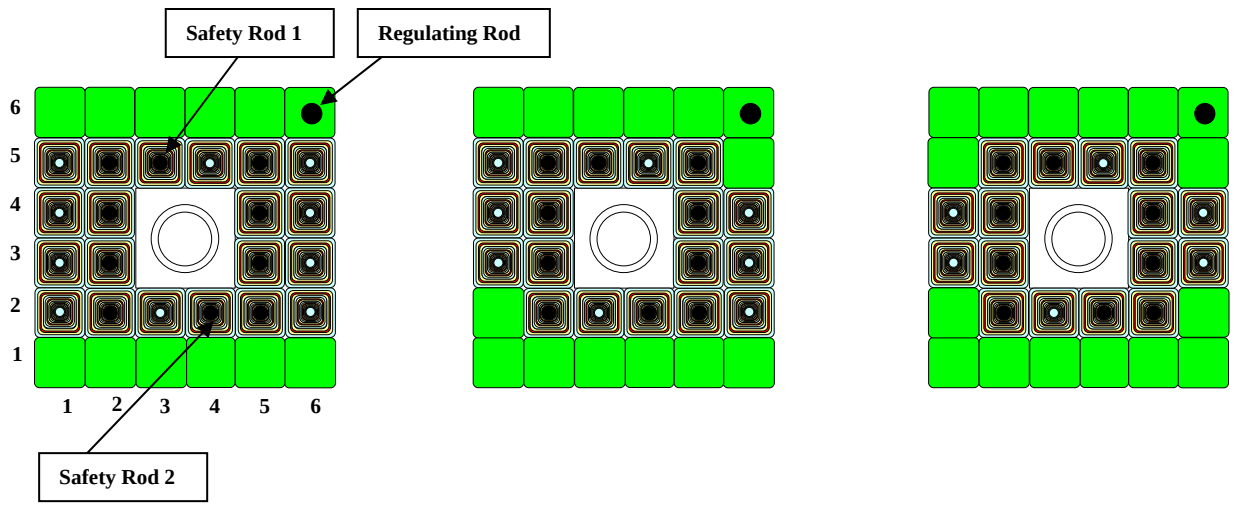
The maximum values of radial and axial peaking factors which are defined respectively as The ratio of average heat flux of the hot channel to the average heat flux of the channels in the core and the ratio of the maximum heat flux of the hot channel to the average heat flux of the hot channel. These two factor and the total hot channel factor which is the product of the radial and axial factors are tabulated in Table(4) with all control rods are fully raised and with critical system cases.

Table(64) Maximum values of radial and axial peaking factors

	Conditions	FR		FZ		FN		Hot channel	
		HEU	LEU	HEU	LEU	HEU	LEU	HEU	LEU
Case (A)	All CR raised	1.131	1.025	1.29	1.265	1.459	1.297	4-5	4-4
	Critical state	1.210	1.190	1.325	1.256	1.603	1.495	3-2	4-2
Case (B)	All CR raised		1.096		1.278		1.401		4-5
	Critical state		1.254		1.265		1.586		3-2

CALCULATIONS WITH CENTRAL NEUTRON TRAP

The central neutron trap is loaded to the core center with core configuration of 20 HEU fuel assemblies 20, 18, and 16 LEU fuel assemblies as shown in Figure(2) were studied. Results are presented for thee cases, with central trap plugged case(I), only the small plug is replaced by water case(II), and both plugs are replaced by water case(III). Core excess reactivity for these different cases are given in Table(5-a) to Table(5-d). Thermal and fast flux distributions along the central neutron trap are plotted if Figure(3).



(a) HEU or LEU core with 20 fuel assemblies **(b) LEU core with 18 fuel assemblies** **(c) LEU core with 16 fuel assemblies**

Figure(2) core configurations with central Neutron Trap

Table(5-a) Results with Central Neutron Trap and HEU core

	Both plugs of CNT are in place		Only Inner plug is replaced by water		Both plugs are replaced by water	
	k_{eff}	$\Delta k/k$ %	k_{eff}	$\Delta k/k$ %	k_{eff}	$\Delta k/k$ %
Only Safety Rods are raised	0.9669960	-3.413	0.9582840	-4.353	0.9362858	-6.805
All Rods are Raised	1.1499053	13.036	1.1382304	12.144	1.1069747	9.664
All Rods are Inserted	0.9284062	-7.711	0.9219907	-8.461	0.9031587	-10.723

Table(5-b) Central Neutron Trap with 20 LEU Fuel assemblies

	Both plugs of CNT are in place		Only Inner plug is replaced by water		Both plugs are replaced by water	
	k_{eff}	$\Delta k/k$ %	k_{eff}	$\Delta k/k$ %	k_{eff}	$\Delta k/k$ %
Only Safety Rods are raised	1.0196112	+1.923	1.0213250	+2.088	0.9873876	-1.277
All Rods are Raised	1.2032058	16.889	1.1932414	16.186	1.1575305	13.609
All Rods are Inserted	0.9794536	-2.098	0.9863250	-1.386	0.9526536	-4.970

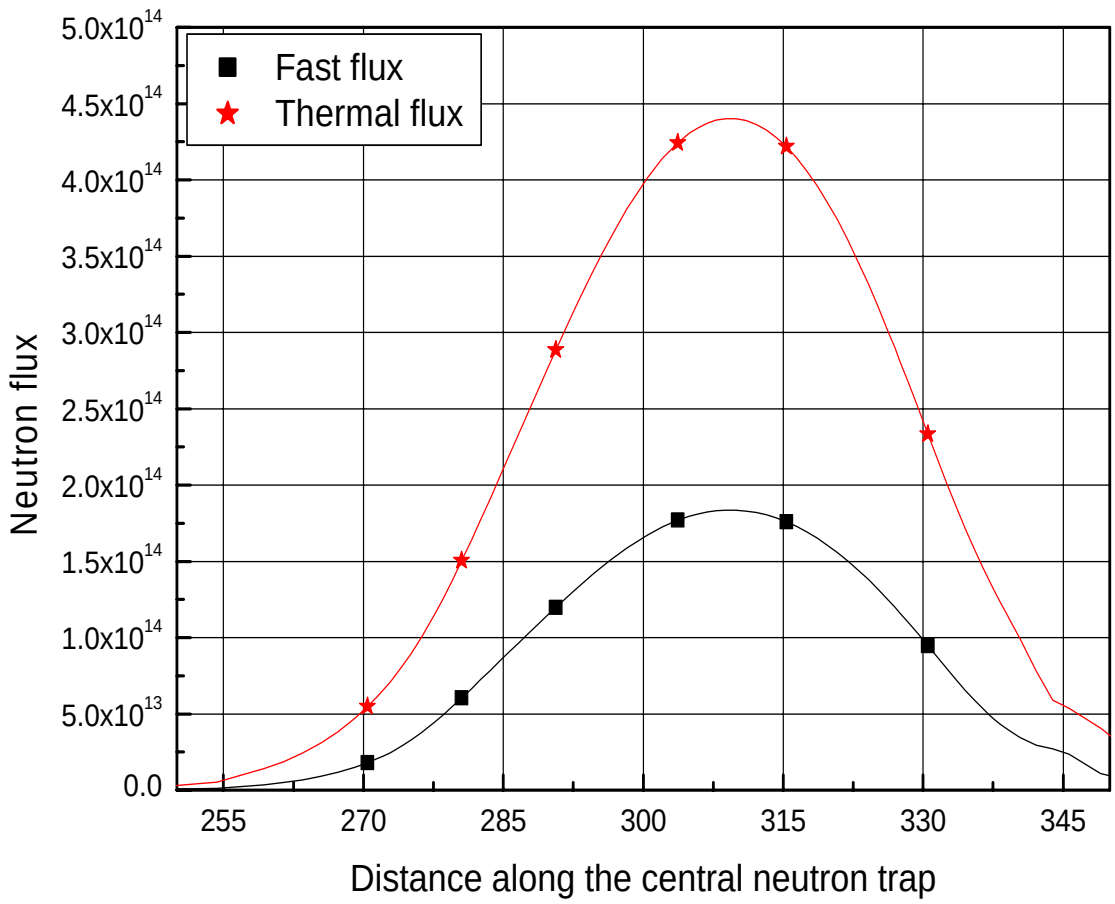
Table(5-c) Central Neutron Trap with 18 LEU Fuel assemblies

	Both plugs of CNT are in place		Only Inner plug is replaced by water		Both plugs are replaced by water	
	k_{eff}	$\Delta k/k$ %	k_{eff}	$\Delta k/k$ %	k_{eff}	$\Delta k/k$ %
Only Safety Rods are raised	0.9993156	-0.0685	0.9979826	-0.20589	0.9653074	-3.5939
All Rods are Raised	1.188686	15.8735	1.1768816	15.02969	1.1406595	12.33142
All Rods are Inserted	0.9554169	-4.7970	0.9585893	-4.319963	0.9269292	-7.88310

Table(5-d) Central Neutron Trap with 16 LEU Fuel assemblies

	Both plugs of CNT are in place		Only Inner plug is replaced by water		Both plugs are replaced by water	
	k_{eff}	$\Delta k/k$ %	k_{eff}	$\Delta k/k$ %	k_{eff}	$\Delta k/k$ %
Only Safety Rods are raised	0.9401636	-6.364	0.9654594	-3.578	0.9332294	-7.155
All Rods are Raised	1.1700999	14.537	1.1559820	13.493	1.1188203	10.620
All Rods are Inserted	0.9229438	-8.349	0.9223619	-8.417	0.8916621	-12.150

The shape of the axial flux distribution along the central neutron trap is given in Figure (3). The maximum values of the fast and thermal neutron fluxes corresponding to a power of 10 MW are given in Table(6) for different cases.



Figure(3) 20 HEU FA Fast and thermal flux distributions along the central neutron trap

Table(6) k_{eff} **and maximum flux values for different core loadings with central neutron trap**

Case	k_{eff}	ϕ_{fast_max}	ϕ_{th_max}
20 FA HEU core; only the small plug is replaced by water	1.0005633	2.14199E14	1.63944E14
20 FA HEU core; both plugs are replaced by water	1.0000528	1.78495E14	1.75405E14
20 FA LEU core; only the small plug is replaced by water	The system is supper critical with only safety rods raised		
20 FA LEU core; both plugs are replaced by water	1.0007757	1.371732E14	1.69387E14
18 FA LEU core; only the small plug is replaced by water	1.0001284	2.417355E14	1.6791E14
18 FA LEU core; both plugs are replaced by water	1.0008997	2.067008E14	1.91337E14
16 FA LEU core; only the small plug is replaced by water	1.0008346	2.807950E14	1.95459E14
16 FA LEU core; both plugs are replaced by water	1.0009155	2.298468E14	2.13172E14

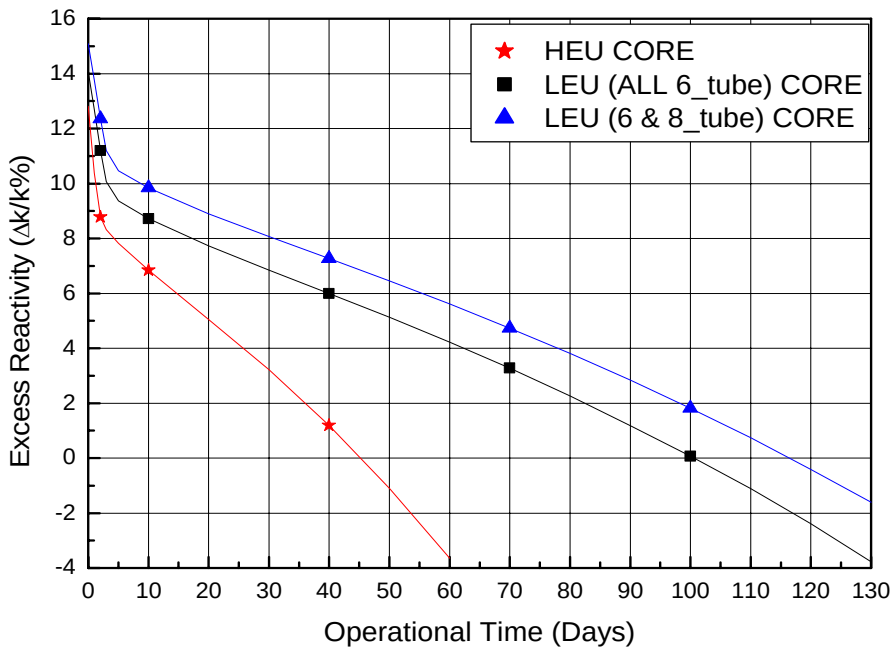
Control rods are calibrated with central neutron trap at critical system. The worth of all rods with HEU and LEU cores are tabulated in Table(7).

TablE(7) CONTROL RODS WORTH WITH CENTRAL NEUTRON TRAP

Rod	***20 FA of HEU*** core		***18 FA of 8 & 6-tube type LEU*** core	
	Only small plug removed $\Delta k/k\,\%$	Both plugs removed $\Delta k/k\,\%$	Only small plug removed $\Delta k/k\,\%$	Both plugs removed $\Delta k/k\,\%$
Safety Rod 1	1.6540	1.6737	1.8921	1.7845
Safety Rod 2	1.6975	1.9198	1.9234	1.9733
Regulating R	0.4694	0.5273	0.3357	0.5002
Shim Rod 1	2.7516	2.5701	2.6221	2.5898
Shim Rod 2	2.8285	2.6845	2.5219	2.5186
Shim Rod 3	2.0537	1.6854	2.4156	2.0408
Shim Rod 4	2.2740	1.9459	2.0780	1.9008
Shim Rod 5	1.1873	1.1804	1.4111	1.3519
Shim Rod 6	1.6387	1.6610	2.1273	2.0800
Shim Rod 7	1.9195	2.0668	1.9492	2.1602
Shim Rod 8	2.5324	2.3262	2.1879	2.4525

BURN UP CALCULATIONS

Burn up calculations starting with fresh fuel is performed at hot conditions (fuel, clad, and coolant temperatures are taken as $380\ ^0K$, $360\ ^0K$ and $325\ ^0K$ respectively). The results are obtained for both HEU and LEU cores and with all control rods are fully raised. The excess reactivity is plotted against operational time in Figure(4).



Figure(4) Burn up for fresh LEU cores

CONCLUSIONS

Changing the IRT-1 reactor fuel from IRT-2M highly enriched uranium to IRT-4M low enriched uranium results in increasing the core excess reactivity by1% and decreasing the total worth of the control rods by 1%. The change in the integral worth of the individual rods is given in Table(3). As shown in Table(3) the ratio of *LEU* maximum flux values to *HEU* values in the fuel region in the axial direction increases by a factor of 1.1 in the fast energy range and decreases by 0.6 in the thermal energy range.

It is very clear that the methods and tools used in the calculations gave very good results. According to the obtained results there is no contradictions related to criticality and control rods worth in converting the reactor fuel from IRT-2M (*HEU*) to IRT-4M (*LEU*) fuel.

Investigations with central neutron trap show that the thermal neutron flux can be increased in the core almost three times. Burn up analysis, as shown in Figure(4) for an excess reactivity of 1% as the end of the cycle reactivity the first cycle length is about 45 days for HEU core, about 100 days for the 6-tube LEU core and about 117 days for the mixed 6 and 8-tube LEU core

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