RERTR 2009 — 31st INTERNATIONAL MEETING ON
REDUCED ENRICHMENT FOR RESEARCH AND TEST REACTORS

November 1-5, 2009
Kempinski Hotel Beijing Lufthansa Center
Beijing, China

FUEL CYCLE ANALYSES OF THE KIPT NEUTRON SOURCE FACILITY WITH LOW ENRICHED URANIUM FUEL

Zhaopeng Zhong and Yousry Gohar
Nuclear Engineering Division
Argonne National Laboratory
9700 South Cass Avenue, Argonne, IL. 60439 USA

ABSTRACT

Argonne National Laboratory (ANL) of USA and Kharkov Institute of Physics and Technology (KIPT) of Ukraine have been collaborating on the conceptual design development of a neutron source facility consisting of electron accelerator driven subcritical assembly. The neutron source driving the subcritical assembly is generated from the interaction of 100 KW electron beam with a natural uranium target. The subcritical assembly surrounding the target is fueled with low enriched WWR-M2 type hexagonal fuel assemblies. The U-235 enrichment of the fuel material is < 20%. The facility will be utilized for basic and applied research, producing medical isotopes, and training young specialists. With the 100 KW electron beam power, the total thermal power of the facility is ~360 kW including the fission power of ~260 kW. The burnup of the fissile materials and the buildup of fission products continuously reduce the system reactivity during the operation, decrease the neutron flux level, and consequently impact the facility performance. To preserve the neutron flux level during the operation, the fuel assemblies should be added and shuffled for compensating the lost reactivity caused by burnup. This paper studies the fuel cycle and shuffling schemes of the KIPT ADS to preserve the system reactivity and neutron flux level during the operation.

1.0 Introduction

National and international research institutes are considering accelerator driven systems (ADS) in their fuel cycle scenarios for transmuting actinides and long-lived fission products. Therefore, several studies and experiments have been performed using accelerator driven subcritical systems. As a part of the collaboration activity between the United States of America and Ukraine, Argonne National Laboratory (ANL) and the National Science Center-Kharkov Institute of Physics and Technology (NSC-KIPT) have been collaborating on developing a neutron source facility based on the use of electron accelerator driven subcritical system [1]. The main functions of this facility are the medical isotope production and the support of the Ukraine nuclear industry. Physics experiments and material research will also be carried out utilizing the sub-critical assembly. KIPT did have a plan to construct this facility using high-enriched uranium (HEU) fuel. These collaborative studies showed that the use of low enriched uranium (LEU) instead of HEU enhances the facility performance and the main system choices and design parameters were defined [2].

The subcritical assembly operates with 100 KW electron beam using electron energy in the range of 100 MeV to 200 MeV. When the core is loaded with 35 fuel assemblies, the total generated thermal power is ~360 KW for the 100 MeV electrons. The total fission power of the system is about 260 kW and the neutron flux level is $\sim 10^{13}$ $n/s \cdot cm^2$. The average power density in the fuel is ~70 W/cm^3. The reactivity and flux level of the subcritical assembly decrease during the operation due to the fuel burnup and the buildup of fission products. The fuels of the subcritical assembly should be shuffled or new fuel assemblies should be added to increase the reactivity and preserve the neutron flux level. The k_{eff} of the subcritical core during the depletion should be kept ~0.98 or lower to satisfy the design requirements.

Unlike the critical power reactors, the ADS facility doesn't use control rods or burnable poison materials to control its reactivity. Therefore additional reactivity should be introduced to the ADS frequently during the operation to compensate the lost reactivity and preserve the neutron flux level caused by the burnup of the fissile isotopes and the accumulation of the fission products. In addition during operation, the Xe-135 concentration impacts the subcritical assembly performance. At the beginning of burnup cycle, the Xe-135 accumulates very fast and reduces the effective multiplication factor of the subcritical assembly by ~1000 pcm after two days of operation due to its large thermal absorption cross section. Such change reduces the neutron flux and the subcritical assembly needs additional reactivity for compensation. However after shutdown, the Xe-135 decays quickly and the lost reactivity is recovered. Such increase can results in neutron multiplication factor above 0.98, which is not desirable. To avoid this situation, the reactivity changes (gain or loss) should be limited, which can be accomplished by changing the position of the fuel assemblies.

2. Geometrical Model and Calculational Methodology

The conceptual design of KIPT ADS has defined the geometry of the subcritical assembly, the target assembly, and its location for producing the neutron source, the fuel loading, the reflector material and its thickness, and the main design parameters [2]. The fuel design is WWR-M2 type [3], which is used for Kiev research reactor and other test reactors with water coolant. It has a hexagonal geometry with 3.5 cm pitch. The fuel assembly has uranium oxide material in an aluminum matrix, aluminum clad, and 50 cm active length. The U-235 enrichment is less than 20%. The subcritical core is initially loaded with 35 fuel assemblies, surrounded by graphite reflector inside a water tank. The natural uranium target has a total length of 7.9 cm and the target material length is divided into 12 separate plates with water coolant on the both sides of each plate. The thermal hydraulics and the structural analyses defined the thickness of each plate to limit the maximum temperature and the maximum induced thermal stress of the target uranium material. The electron interactions with the uranium target produce high energy photons, which generate neutrons through photonuclear reactions. Such interactions occur at the assembly center and the produced neutrons drive the subcritical assembly. The radial configuration of the subcritical assembly is shown in Figure 1, which includes the target, the fuel assemblies, and the graphite reflector blocks.

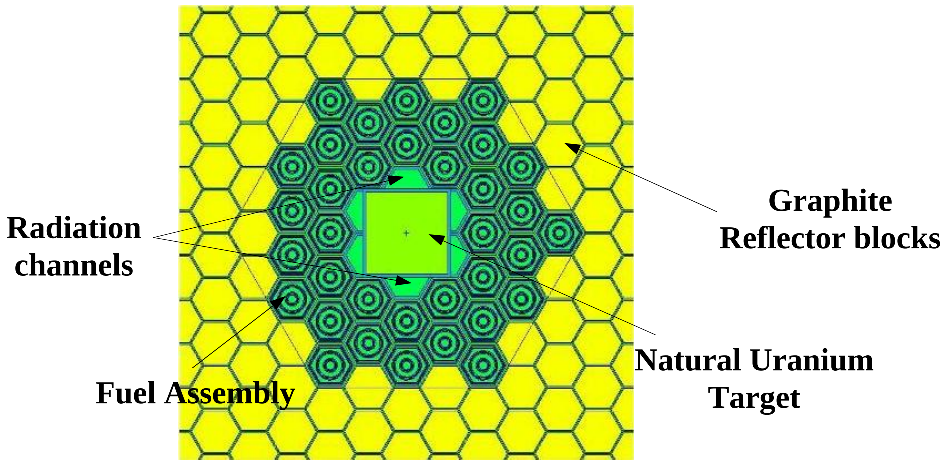

Figure 1. Radial configuration of the subcritical assembly

The coupling procedure of MCNPX [4] and MCB [5, 6] computer codes has been introduced [7, 8] and utilized in this analysis. First, a MCNPX calculation is performed starting with 100 MeV electrons and the neutron fission events are turned off to generate a neutron source file. This file includes the position, the energy, the direction vector, and the weight of each generated neutron inside the target material from the photonuclear reactions caused by the electrons. The file generation utilized the user defined subroutine TALLYX of MCNPX. Then the neutron source file is used by MCB through the user defined subroutine SOURCE driving the subcritical assembly. In this procedure, the neutron yield from electrons is simulated correctly in MCB calculation, although no electron transport involved. The neutron source is normalized to the 100 kW electron beam power. The corresponding neutron source strength in MCB calculation is ~3.11×10^{14} per second and it is assumed to be constant during all the burnup cycles.

In the MCB calculation, the 35 fuel assemblies are divided into three radial burnup zones, based on their radial positions relative to the target, as shown in Figure 2. Each radial zone is divided into five equally spaced axial zones with 10 cm axial length of each zone. In the target, each plate is treated as a separate zone. The total number of burnup zones is twenty seven in the MCB model.

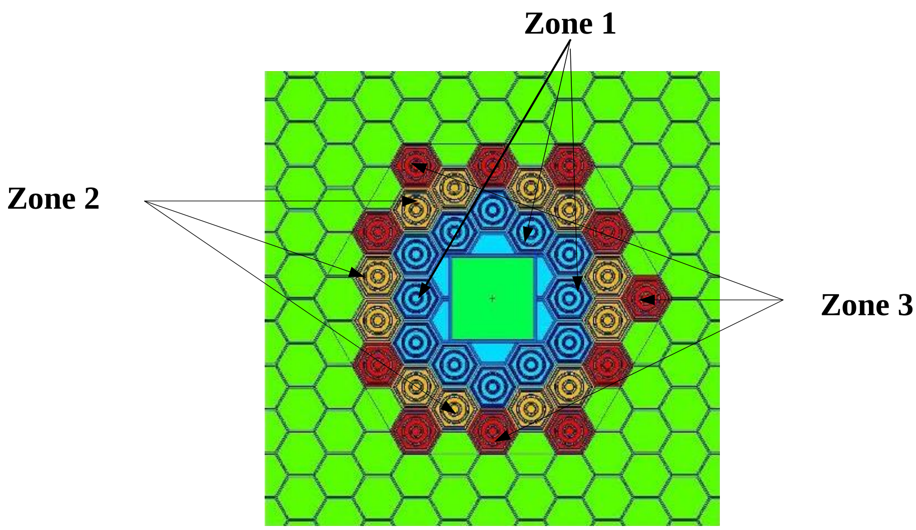

Figure 2. Radial depletion zones for the ADS core

3. Reactivity Compensation

The whole burnup cycle is divided into multiple steps for adding reactivity into the subcritical core, to compensate the losses caused by the burnup of fissile material and buildup of fission products for each burnup step. To reduce the neutron flux change in the subcritical during operation, the reactivity loss at each burnup step should be about 200 to 300 pcm. Based on the previous study [7, 8], adding one fresh fuel assembly increases k_{eff} by ~700 pcm, which could overcompensate the reactivity loss and can results in k_{eff} value greater than 0.98, which represents an operation concern. To avoid this problem, core configuration shuffling has been analyzed, the objective is to adding 200 ~ 300 pcm reactivity each shuffling.

In order to increase the effective multiplication factor of the subcritical assembly, a series of MCNPX calculation has been performed, starting from the initial configuration loaded with 35 fuel assemblies, where the fuel assemblies are moved outward in the radial direction in stepwise. The first step moved only two peripheral fuel assemblies as shown in Fig. 3. The configuration of each calculation and the corresponding K_{eff} are shown in Figure 3. Fresh fuel assemblies are used in all these calculations, therefore the increase of k_{eff} value is only due to the additional neutron moderation from the additional water between the fuel assemblies. These results show that each step increases the k_{eff} ff value by ~300 pcm from the previous step. Such rearrangement of the fuel assemblies are used to compensate the reactivity loss during the burnup. The maximum increase of k_{eff} from the eight steps is ~1800 pcm relative to the original configuration.

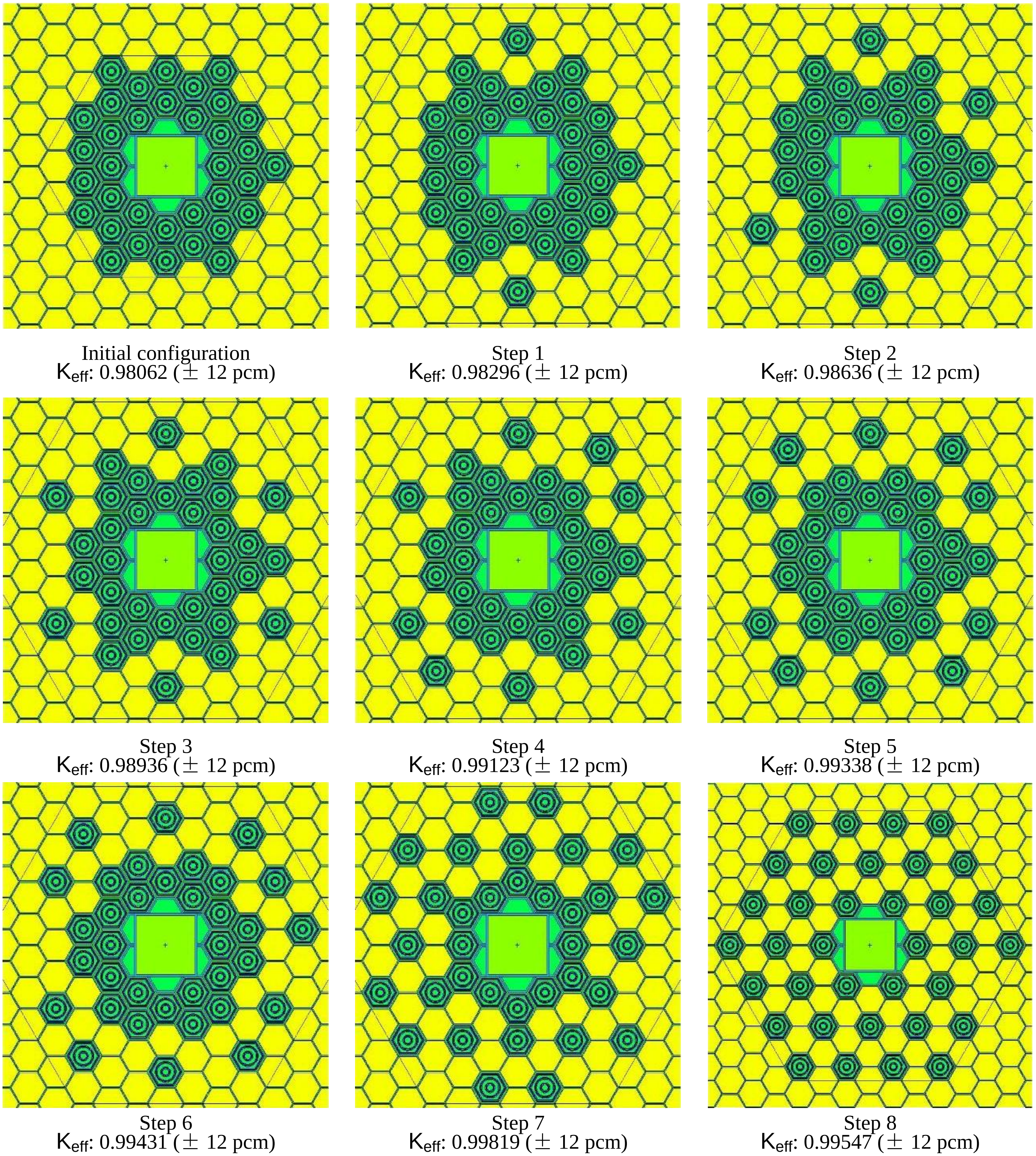

Figure 3. Effective neutron multiplication values of different configurations with 35 fresh fuel assemblies

4. Burnup analyses of the subcritical assembly based on fuel shuffling scheme

The burnup behavior of the subcritical assembly with the fuel shuffling scheme has been studied, based on the computation model and reactivity compensation scheme discussed in previous sections. Monte Carlo code MCB [5] is selected to perform the burnup calculation with the neutron source prepared by MCNPX [4]. The whole burnup cycle is divided into steps to employ the fuel shuffling scheme. The basic principle of the reactivity compensation scheme is to spread the fuel assemblies to increase the neutron moderation for enhancing the neutron multiplication.

It is known from the previous study [7, 8] that the accumulation of the fission products, especially Xe-135, reduces the effective neutron multiplication factor by ~900 pcm after two days, although, the change in the fissile material loading is very small during these two days. After the facility is shut down, Xe-135 decays quickly and the lost reactivity recovered in a few days. The impact of the accumulation and the decay of fission products on the reactivity during the operation and after the shutdown as a function of the time are shown in Figure 4 for the subcritical assembly with 35 fresh fuel assemblies and two days of operation.

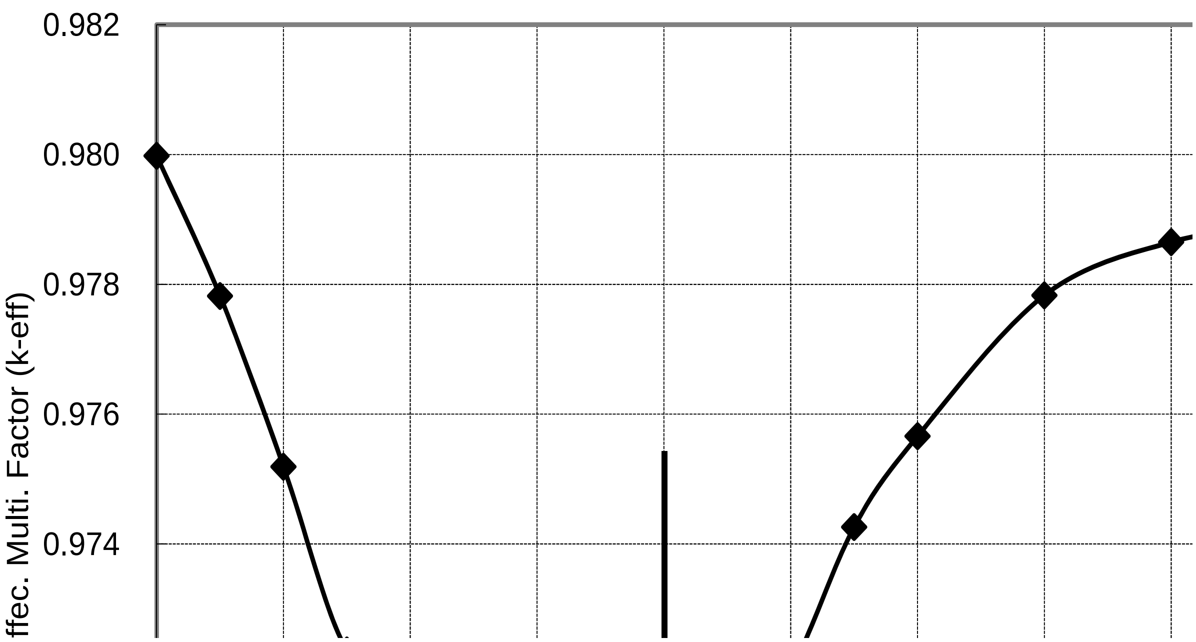

Figure 4. K_{eff} value during the first two days of operation and after the shutdown of the subcritical assembly with 35 fresh fuel assemblies

To compensate the reactivity drop caused by the fission products, mainly Xe-135, one fresh fuel assembly is added after two days of burnup. Then the assembly is made sparser by moving two fuel assemblies outward step by step. For the first burnup cycle with the subcritical core initially loaded with 35 fresh fuel assemblies, the core configurations at each depletion step are shown in Figure 5, and the Keff history is shown

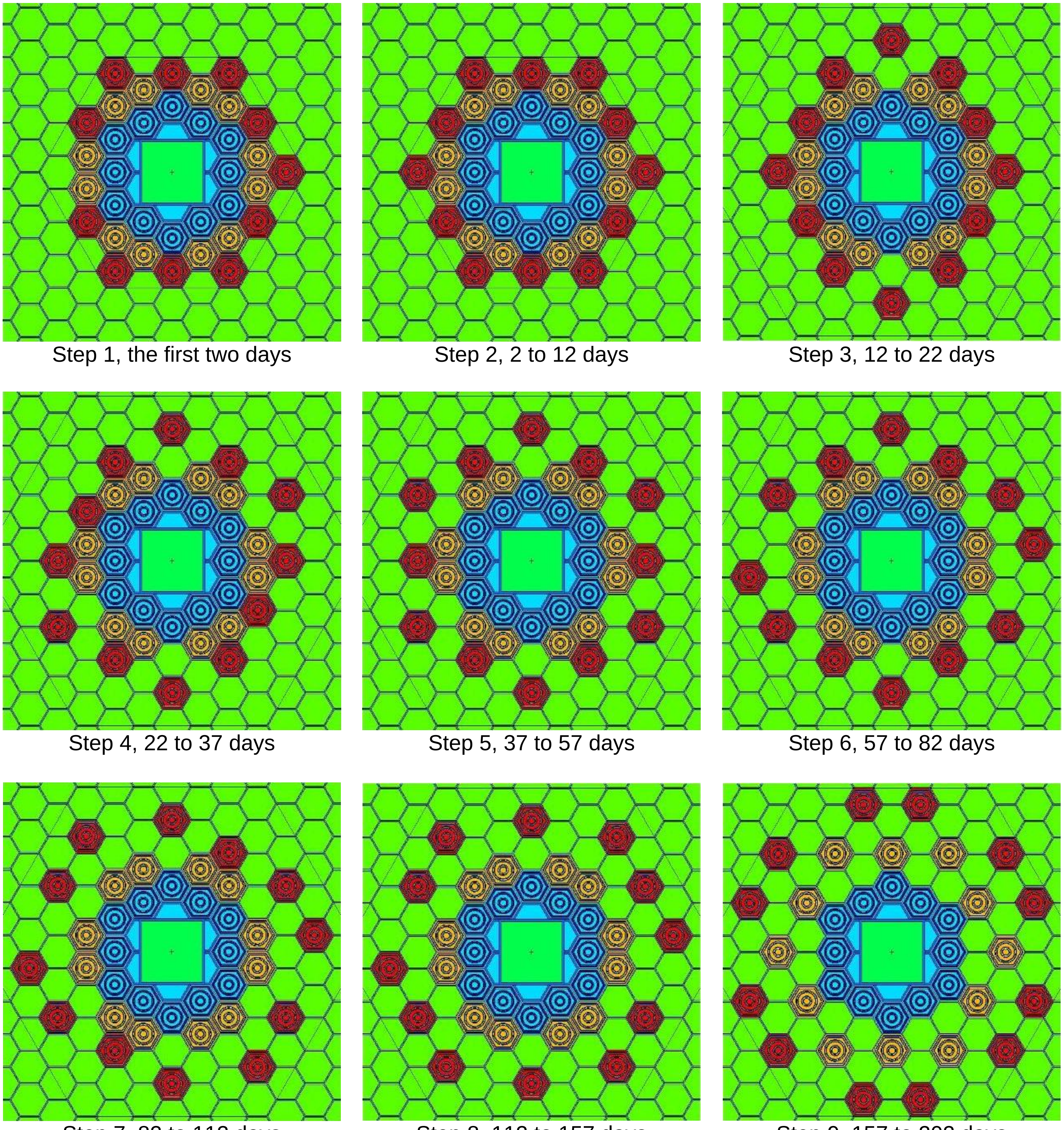

Figure 5. Subcritical assembly configuration for each burnup step of the first burnup cycle, with one fresh fuel assembly added at step 2

in Figure 6. Two hours is assumed between the steps to move the two fuel assemblies since the operation of the fuel machine does not require cooling time after shutdown. Using this fuel shuffling scheme, k_{eff} is kept in the range of 0.975 to 0.980, therefore the neutron flux level can be preserved during operation as shown in Error! Reference source not found.**. The neutron flux in the radiation channels at the beginning and the end of each depletion step is also shown in Table I. The burnup of each radial fuel zone at the end of each depletion step is shown in**
Table II. At the end of the first burnup cycle, 202 days of burnup, the minimum U-235 enrichment is ~19%.

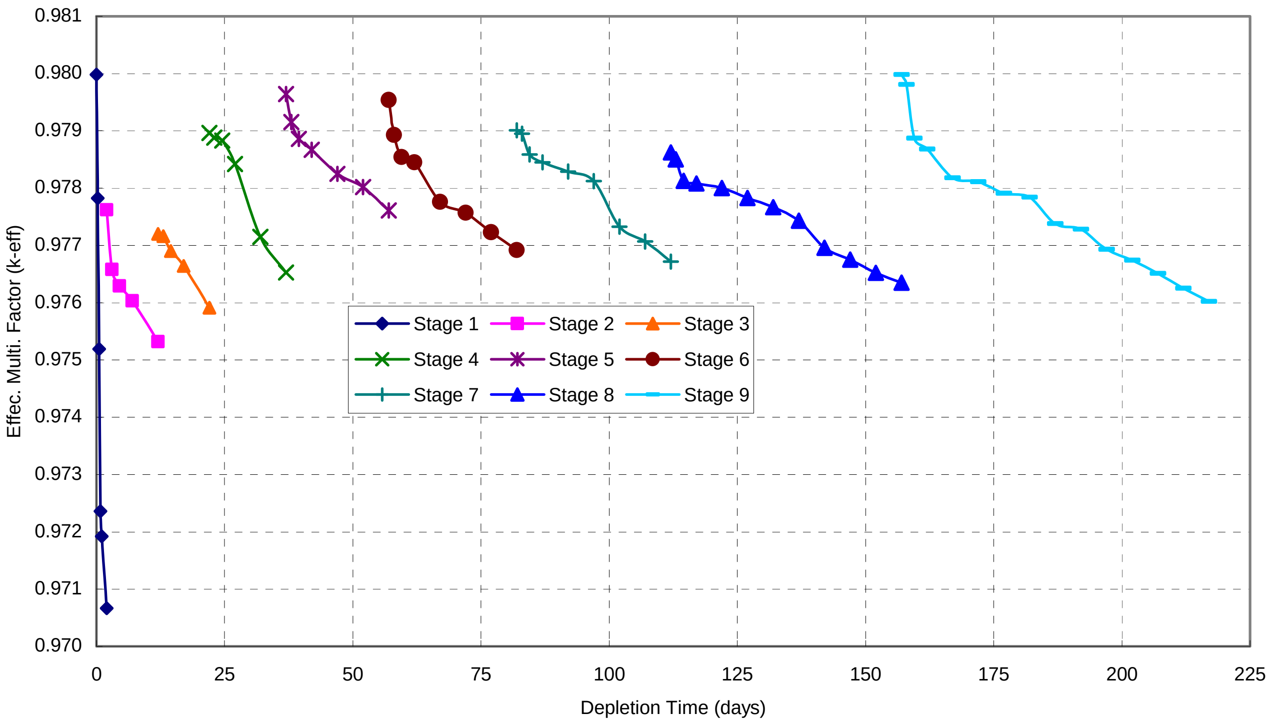

Figure 6. Keff during the first burnup cycle, with one fresh fuel assembly added at the beginning of step 2

Although using this fuel shuffling scheme can preserve the reactivity and neutron flux level, the decay of fission products will increase the reactivity after shutting down the facility. The decay behavior of the subcritical at the beginning and end of step 9 has been analyzed, because the highest k_{eff} value during the first burnup cycle appears at the beginning of step 9. With the facility shut down at the end of step 9, the k_{eff} can reach 0.986 after a few days of decay. If the facility has to be shut down at the beginning of step 9 unexpectedly, the k_{eff} can reach ~0.99 because of the decay of the fission products decay. To reduce the reactivity after shutting down the facility, one fuel assembly in the peripheral region should be removed. In this way the system K_{eff} can be reduced ~900 pcm, and is kept ~0.98 or lower. The decay behavior of the subcritical core at the beginning and end of step 9 is shown in

Figure **7**. However, the better approach is to avoid adding the fresh fuel assembly after the first step (two days of operation) and to start moving the fuel assemblies outward to get the necessary reactivity increase. This approach insures that the reactivity will not exceed 0.98 after the decay of the fission products.

Table I. Keff and neutron flux values at the beginning and end of each step, for 36 fuel assemblies loaded in the first burnup cycle, calculated by MCB

Step	Beginning of the step			End of the step		
	Keff	Neutron flux along the fuel(n/cm^2.s)	Neutron flux along the target(n/cm^2.s)	Keff	Neutron flux along the fuel(n/cm^2.s)	Neutron flux along the target(n/cm^2.s)
1	0.97998 (±36 pcm)	2.407e+13 (±1.63%)	2.961e+13 (±1.58%)	0.97067 (±33 pcm)	1.705e+13 (±1.36%)	2.133e+13 (±1.31%)
2	0.97762 (±36 pcm)	2.116e+13 (±1.52%)	2.619e+13 (±1.46%)	0.97532 (±32 pcm)	1.987e+13 (±1.50%)	2.469e+13 (±1.45%)
3	0.97720 (±32 pcm)	2.134e+13 (±1.52%)	2.638e+13 (±1.46%)	0.97591 (±32 pcm)	2.037e+13 (±1.50%)	2.528e+13 (±1.45%)
4	0.97896 (±28 pcm)	2.209e+13 (±1.55%)	2.724e+13 (±1.50%)	0.97653 (±30 pcm)	2.192e+13 (±1.56%)	2.704e+13 (±1.51%)
5	0.97964 (±36 pcm)	2.346e+13 (±1.60%)	2.881e+13 (±1.55%)	0.97761 (±39 pcm)	2.241e+13 (±1.54%)	2.756e+13 (±1.49%)
6	0.97954 (±31 pcm)	2.423e+13 (±1.60%)	2.972e+13 (±1.55%)	0.97692 (±35 pcm)	2.126e+13 (±1.56%)	2.624e+13 (±1.50%)
7	0.97901 (±24 pcm)	2.252e+13 (±1.61%)	2.768e+13 (±1.55%)	0.97672 (±32 pcm)	2.083e+13 (±1.47%)	2.572e+13 (±1.42%)
8	0.97862 (±32 pcm)	2.248e+13 (±1.57%)	2.765e+13 (±1.52%)	0.97635 (±32 pcm)	2.073e+13 (±1.53%)	2.556e+13 (±1.48%)
9	0.97998 (±38 pcm)	2.281e+13 (±1.56%)	2.792e+13 (±1.51%)	0.97602 (±32 pcm)	1.969e+13 (±1.47%)	2.423e+13 (±1.41%)

Table II Burnup of fuel assemblies at the end of different steps, with 36 fuel assemblies loaded in the first burnup cycle

Steps	Burnup Step Days	Total Burnup Days	Burnup (kWD/kgU)			
			Zone 1	Zone 2	Zon3 3	Averag
1	2	2	40.18	41.76	48.72	43.55
2	10	12	241.94	248.14	293.14	261.07
3	10	22	460.06	460.58	549.10	489.91
4	15	37	790.42	781.62	937.78	836.61
5	20	57	1236.10	1214.40	1465.80	1305.43
6	25	82	1792.40	1764.16	2127.40	1894.65
7	30	112	2459.30	2433.98	2923.48	2605.59
8	45	157	3440.34	3434.56	4098.44	3657.78
9	60	217	4823.94	4859.56	5601.64	5095.05

At the end of the first burnup cycle, it is difficult to get additional reactivity by moving the fuel assemblies. However, the fuel assemblies do have ~19% U-235 enrichment and can be utilized for further operation. New fuel assemblies should be added into the subcritical core with the fuel assemblies loaded compactly. Then the second fuel cycle repeats the fuel shuffling scheme used in the first fuel cycle.

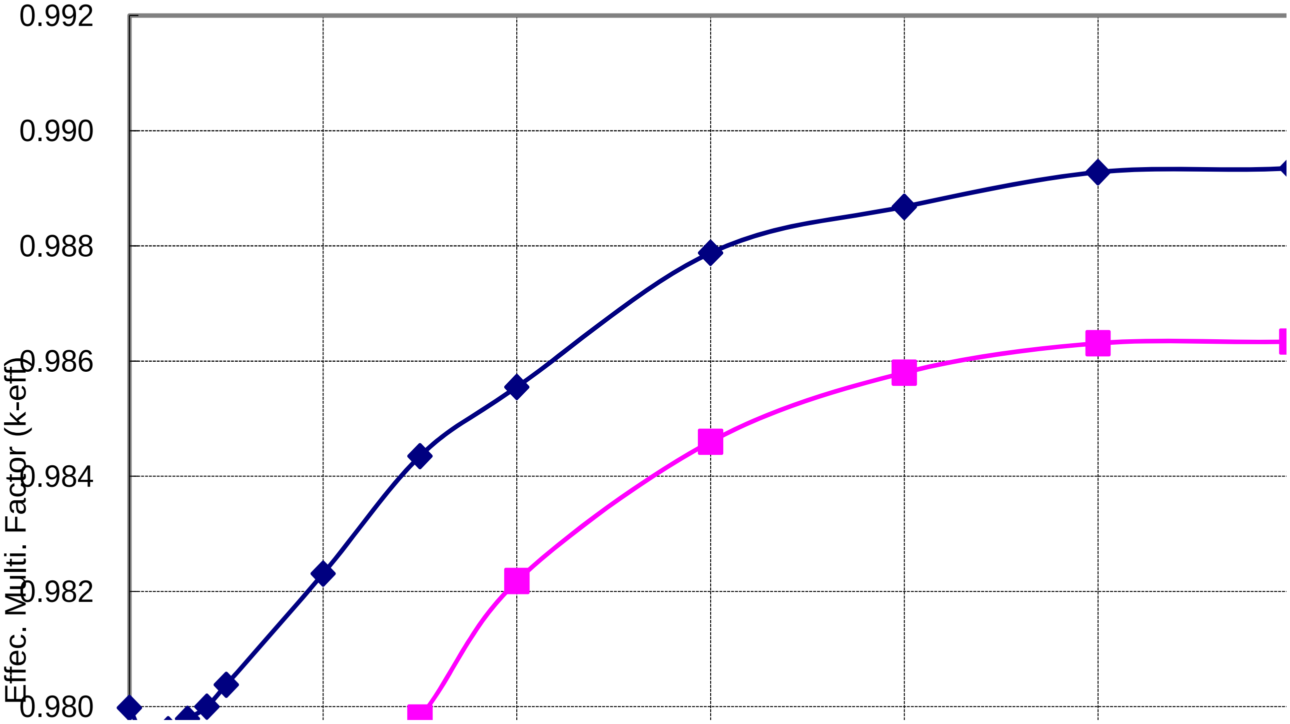

Figure 7. K_{eff} Decay history at the beginning and end of the step number 9

5. Summary and Conclusions

A fuel shuffling scheme of the KIPT accelerated driven subcritical assembly, at different burnup steps, has been analyzed. It is shown that spreading the fuel assemblies away from each other in step wise provides a mechanism for compensating the reactivity loss during the operation. This approach preserves the neutron flux level during the operation. The fuel shuffling scheme of the subcritical assembly, which is initially loaded with 35 fresh fuel assemblies, has been studied in this paper for the first burnup cycle (~200 days). The k_{eff} value of the subcritical system is in the range of 0.975 to 0.980 by performing fuel shuffling at each depletion step. A consideration must be given to the maximum allowed k_{eff} value during the shutdown time between the different burnup stages. This value will define will define the k_{eff} operating range and the achievable neutron flux assuming passive approach to control the neutron multiplication. If an active engineering approach is included after shutdown, the k_{eff} value during operation will define the facility performance.

6. Acknowledgement

This project is supported by the U.S. Department of Energy, Office of Global Nuclear Material Threat Reduction (NA213), National Nuclear Security Administration of US Department of Energy

7. References

[1] Y. Gohar, I. Bolshinsky, D. Naberezhnev, "Accelerator-driven sub-critical facility: Conceptual design development," Nuclear Instrument and Methods in Physics Research A 562 (2006)870-874.

[2] D. Naberezhnev Y. Gohar, J. Bailey, H. Belch, "Physics analyses of an accelerator-driven sub-critical assembly," Nuclear Instrument and Methods in Physics Research A 562 (2006)841-844.

[3] R. B. Pond, et al., Neutronic performance of the WWR-M research reactor in Ukraine, Proceedings of the 24th International Meeting on Reduced Enrichment for Research and Test Reactors, San Carlos de Bariloche, Argentina, 3-8 November 2002.

[4] Denise B. Pelowitz, editor, "MCNPX™ USER'S MANUAL", Los Alamos Report, LA-CP-05-0369, April 2005.

[5] J. Cetnar, W. Gudowski, J. Wallenius, "User Manual for Monte Carlo Continuous Energy Burnup (MCB) Code – Version 1C", NEA-1643/01, August, 2002

[6] A. Talamo, W, Ji, J. Cetnar, W. Gudowski, "Comparison of MCB and MONTEBURNS Monte Carlo burnup codes on a one pass deep burn."Annals of Nuclear Energy 33 (2006), 1176-1188.

[7] Z. Zhong, Y. Gohar, A. Talamo, "Burnup Calculations for KIPT Accelerator Driven Subcritical Facility Using Monte Carlo Computer Codes - MCB and MCNPX ", Argonne Report, ANL-09/07, February 2009

[8] Z. Zhong, Y. Gohar, A. Talamo, "MCNPX and MCB Coupled Methodology for the Burnup Calculation of the KIPT Accelerator Driven Subcritical System", International Conference on Mathematics, Computational Methods & Reactor Physics (M&C 2009), Saratoga Springs, NY, May 2009.