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Burnup Calculations for KIPT Accelerator Driven Subcritical Facility Using Monte Carlo Computer Codes - MCB and MCNPX

Nuclear Engineering Division

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Burnup Calculations for KIPT Accelerator Driven Subcritical Facility Using Monte Carlo Computer Codes - MCB and MCNPX

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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 an electron accelerator driven subcritical (ADS) facility [1], using the KIPT electron accelerator. The neutron source of the subcritical assembly is generated from the interaction of 100 KW electron beam with a natural uranium target [2]. The electron beam has a uniform spatial distribution and electron energy in the range of 100 to 200 MeV. The main functions of the subcritical assembly are the production of medical isotopes and the support of the Ukraine nuclear power industry. Neutron physics experiments and material structure analyses are planned using this facility.

With the 100 KW electron beam power, the total thermal power of the facility is ~375 kW including the fission power of ~260 kW. The burnup of the fissile materials and the buildup of fission products reduce continuously the reactivity during the operation, which reduces the neutron flux level and consequently the facility performance. To preserve the neutron flux level during the operation, fuel assemblies should be added after long operating periods to compensate for the lost reactivity. This process requires accurate prediction of the fuel burnup, the decay behavior of the fission produces, and the introduced reactivity from adding fresh fuel assemblies.

The recent developments of the Monte Carlo computer codes, the high speed capability of the computer processors, and the parallel computation techniques made it possible to perform three-dimensional detailed burnup simulations. A full detailed three-dimensional geometrical model is used for the burnup simulations with continuous energy nuclear data libraries for the transport calculations and 63-multigroup or one group cross sections libraries for the depletion calculations. Monte Carlo Computer code MCNPX and MCB are utilized for this study. MCNPX transports the electrons and the produced neutrons and photons but the current version of MCNPX doesn't support depletion/burnup calculation of the subcritical system with the generated neutron source from the target. MCB can perform neutron transport and burnup calculation for subcritical system using external neutron source, however it cannot perform electron transport calculations. To solve this problem, a hybrid procedure is developed by coupling these two computer codes. The user tally subroutine of MCNPX is developed and utilized to record the information of the each generated neutron from the photonuclear reactions resulted from the electron beam interactions. MCB reads the recorded information of each generated neutron thorough the user source subroutine. In this way, the neutron source generated by electron reactions could be utilized in MCB calculations, without the need for MCB to transport the electrons. Using the source subroutines, MCB could get the external neutron source, which is prepared by MCNPX, and perform depletion calculation for the driven subcritical facility.

I. 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. 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 developmental analyses 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 facility performance parameters [2]. The fuel design is WWR-M2 type, which is used for Kiev research reactor [3] 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 ≤ 20%. The subcritical assembly uses 35 or 36 fuel assemblies surrounded by graphite reflector inside a water tank. The natural uranium target, with the total thickness of ~7.9 cm, is divided into 12 separate plates to have spaces for the required coolant channels. The electron interactions with the target produce high energy photons, which generate neutrons through photonuclear reactions with the target material. 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 reflector assemblies.

The subcritical assembly operates with 100 KW electron beam using either100 MeV or 200 MeV electrons. When the assembly is loaded with 35 fuel assemblies, the total generated thermal power is ~360 KW. The total fission power of the system is about 260 kW and the neutron flux level is ~10^{13} n/s·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. Fresh fuel assemblies should be added to enhance the reactivity and preserve the neutron flux level. Therefore the reactivity of the subcritical system should be predicted accurately to guarantee that the required reactivity value, the neutron flux level, and the performance of the subcritical assembly.

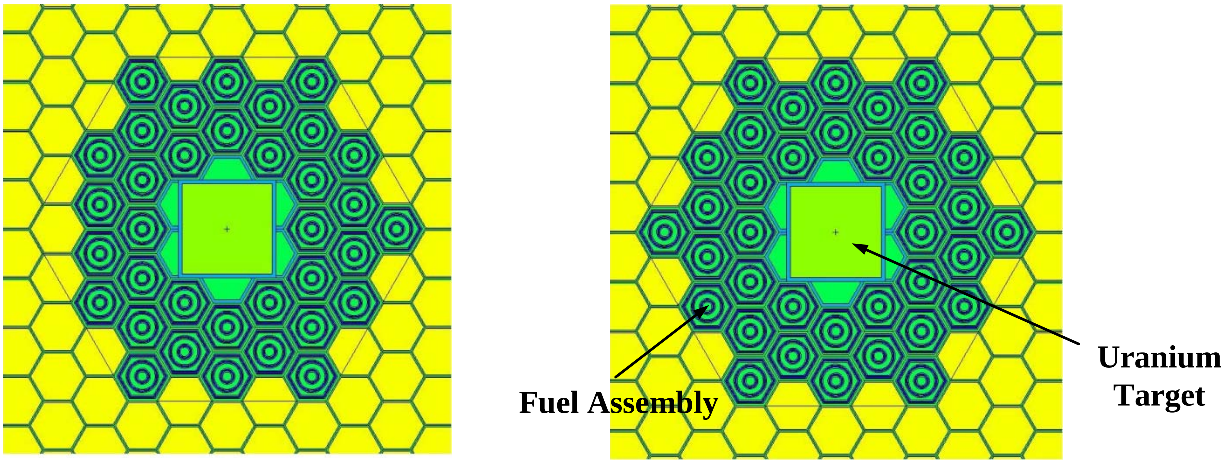


Figure 1. Radial configuration of the subcritical assembly

II. Calculational Methodology

II.1 Computer Codes

The Monte Carlo code – MCNPX [4] has been used in the physics analyses of the KIPT electron accelerator driven subcritical system because of its three dimensional geometrical capability and continuous energy data library for the nuclear data. It has updated capabilities of treating electrons, photons, and neutrons transport, which can be extended into high energy range. In the recent version of MCNPX, a depletion/burnup capability has been implemented, which is based on CINDER90. MCNPX calculates the steady-state neutron flux distributions and CINDER90 uses the neutron flux for depletion calculations and generating new atom densities for the depleted/generated isotopes. MCNPX runs a steady-state calculation to determine the system eigenvalue, the neutron flux in 63-groups structure, reaction rates, and energy deposition. CINDER90 uses the MCNPX-generated values and performs the depletion calculation to generate new number densities at the following time step. MCNPX uses the new number densities and generates another set of neutron flux distributions and reaction rates for CINDER90. This process is repeated for each time step of the defined depletion period [5]. This calculational procedure is different from other Monte Carlo burnup codes, e.g. MONTEBURN, which depend on an interface between independent computer codes using input and output files for each time step. MCNPX is a single executable program, which performs the transport and the depletion calculations in an integrated mode. However, the current version of MCNPX with the depletion/burnup capability is only limited to k-code criticality problem, and it doesn't support depletion calculation for the subcritical system in a fixed source mode. For ADS, if fission source (k-code mode) is used instead of a fixed external source mode, the neutron flux distribution and the neutron spectra could be deformed, which may introduce nontrivial error in the depletion calculation.

MCB is another Monte Carlo Continuous Energy Burnup Code for the general-purpose use to calculate a nuclide density time evolution with burnup or decay [6]. It can be used for eigenvalue calculations and neutron transport of critical and subcritical systems in fixed source mode or k-code mode to obtain reaction rates and energy deposition for burnup calculations. MCB integrates MCNP version 4C [7], which is used for neutron transport calculation, and a novel Transmutation Trajectory Analysis code (TTA) [8]. For burnup calculations, TTA tracks individual fission products and it does not lump the cross sections of the fission products. MCB uses the same input of MCNP with additional input for the depletion/burnup calculations. Complete burnup calculations can be done in a single run with minor modification of an MCNP input file [9]. MCB can perform depletion/burnup calculations for subcritical system in fixed source mode and the neutron flux values and the reaction rates required for the calculations are normalized by the neutron source strength instead of a fixed system power. Since the transport calculation of MCB is based on MCNP4C, it doesn't have the capability to transport electrons in the energy range of 100 MeV to 200 MeV.

II.2 Fixed source burnup calculational method

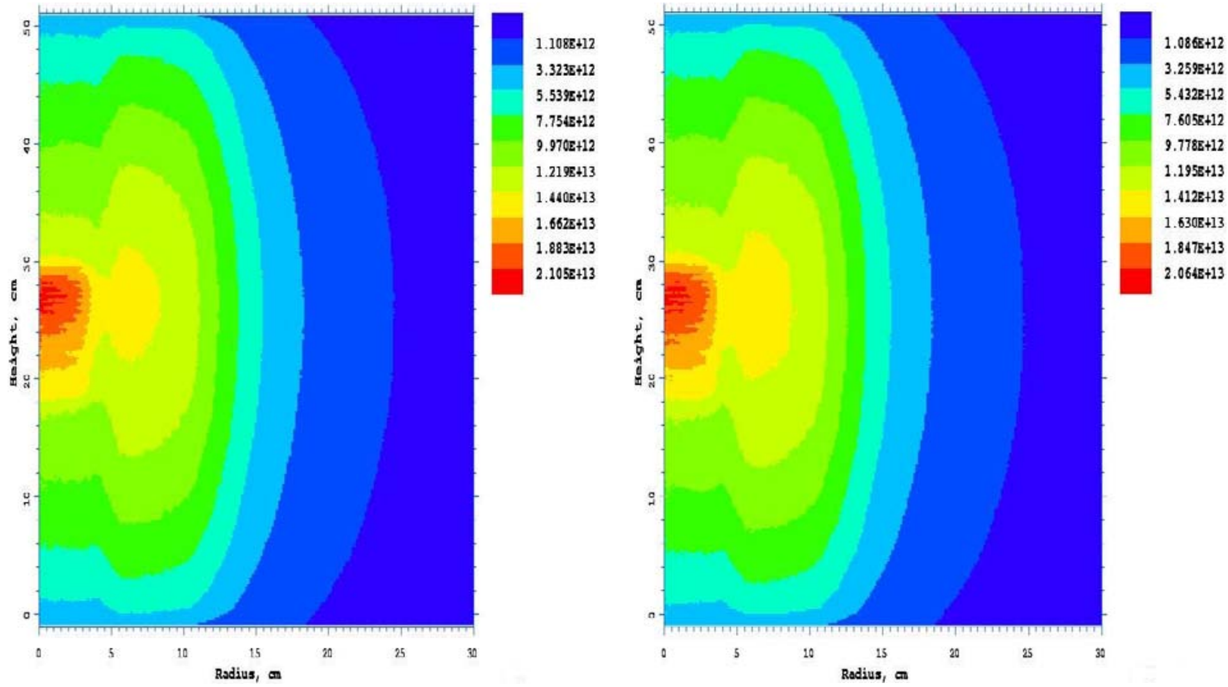
For the depletion calculation of the electron driven subcritical assembly, a hybrid process utilizing MCNPX and MCB is introduced. It is assumed that the neutron yield from the uranium target due to photonuclear reactions initiated by electrons is constant since the target depletion is very small. The change of the uranium density is less than 1% after one full year of operation. Based on this assumption, an external neutrons source is generated, which preserves the location, direction, energy, and weight of each source neutron produced in the target. This neutron source is used to drive the subcritical assembly using MCNPX or MCB Monte Carlo transport codes in fixed source mode calculations. This approach eliminates the need to perform electron and photon transport calculations for every burnup time step. This results in a major computer time save, which make such calculation possible to do.

The TALLYX user-supplied subroutine, which lets users modify any tally, is utilized in MCNPX to generate the external neutron source for driving the subcritical assembly. To generate the neutron source, MCNPX calculation is performed starting from electrons with all neutron fission turned off using NONU card. The TALLYX subroutine is initiated by the F4:n tally card for the target cells associated with FU4 card [10]. Once a neutron appears inside the target cells and its weight is 1.0, its information, energy, direction, position and index of mother electron, will be recorded into a banked source file. Only neutrons with weight equal 1.0 are recorded because. They are newborn neutrons from the target without any collisions. If a neutron undertakes collisions, its weight is reduced or increased based on the collision type and the variance reduction technique. The elimination of the fission events during the calculations insures that all the recorded neutrons are produced directly by electrons instead of neutron fission. Once this file is generated, it contains a lot of duplicated records because neutrons generated in one target plate (cell) can travel into another target plate (cell) without any collisions. Therefore, this banked source file should be processed by another (external) program, removing the duplicated records, which are neutrons with the same energy, direction and mother electron but different spatial position.

In MCNP and MCB, the SOURCE user-specified subroutine, which is used if SDEF, SSR and KCODE cards are absent from the input file, reads the external neutron source file. In MCNPX and MCB calculations, the SOURCE subroutine reads one neutron source record for each starting source neutron. The required number is the NPS values specified in the input file. Each record has the energy, position, direction cosines, and weight of the source neutron. If the NPS value larger than the number of records in the source file, the file is re-read from the beginning to satisfy the requested number of neutrons. The SOURCE subroutine can be used to read the neutron source data for MCB depletion calculation for the subcritical assembly in fixed source mode without the need to perform electron transport calculations. This approach is used to study the burnup of the electron accelerator driven subcritical system without approximations.

II.3 Validation of the fixed source burnup calculational method

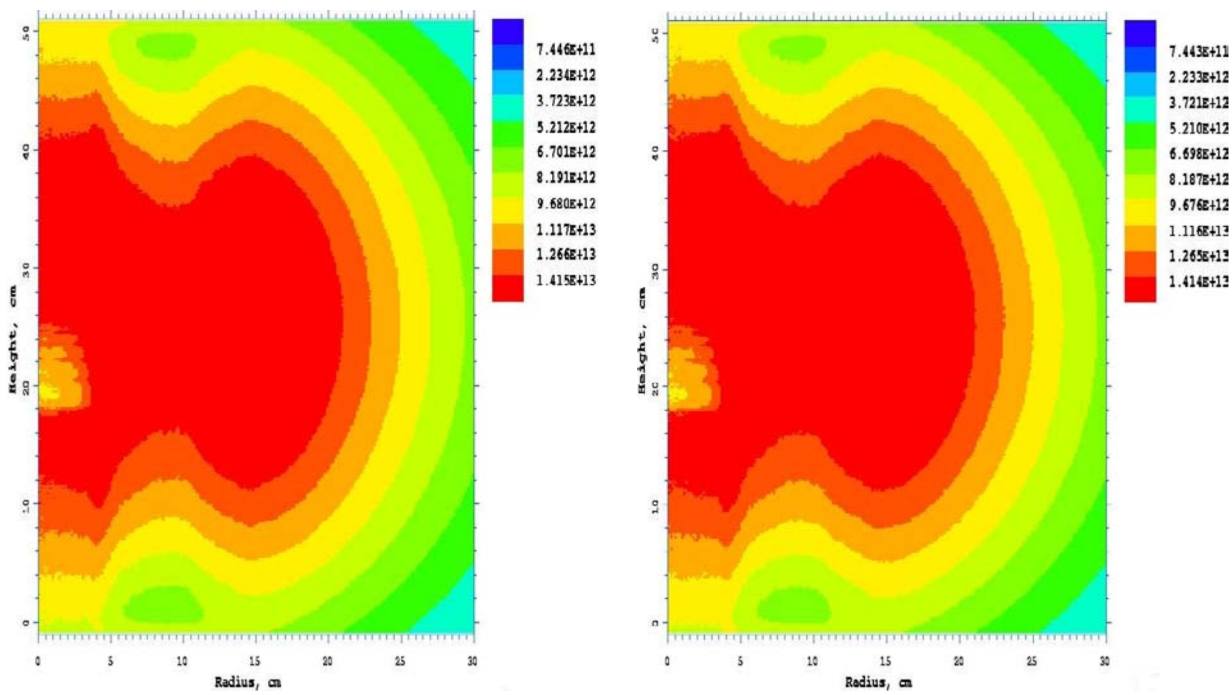
The TALLX and SOURCE subroutines were validated using the subcritical assembly configuration with 35 fuel assemblies, 100 KW electron beam power, and 200 MeV electrons. First, the neutron source file was generated due to one million source electrons. During the neutron source file generation, the fission reactions turned off because the goal is to record only neutrons from photonuclear reactions. The generated file was processed to remove duplicated records, which leaves ~100,000 neutrons recorded into the source file. This number of neutrons is consistent with the expected neutron yield from the photonuclear reactions [11]. The generated file was used in another MCNPX calculation, which has the same geometry and transport only neutrons. This MCNP calculation used 200,000 neutrons, about twice the number of the recorded neutrons in the source file. The spatial neutron flux maps and the energy deposition due to the neutron interactions were calculated. These results are compared with those obtained from the MCNPX calculation using electron source and the results are shown in Figures 2 through 5 and Table I. This comparison shows that the two methods give the same results. Therefore, it is concluded that the TALLX and SOURCE subroutines have been implemented correctly and the neutron source files has the correct number of neutrons.



Neutron source file case

Electron source case

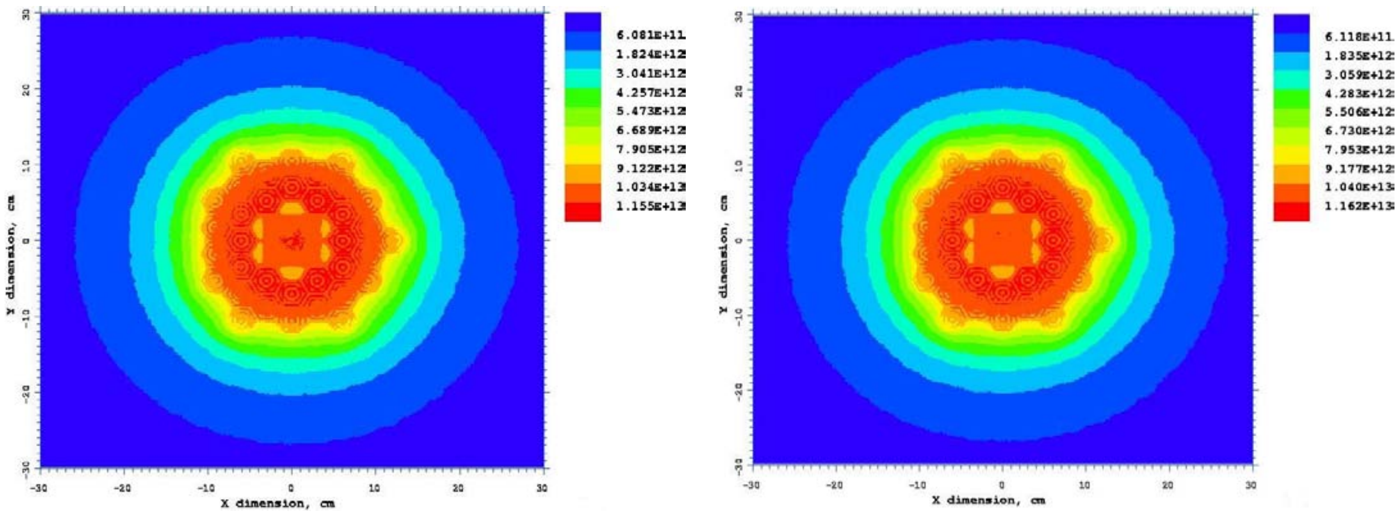
Figure 2. Comparison of R-Z fast neutron flux (E>0.1 MeV) maps calculated using MCNPX with neutron source file and MCNPX transporting 200 MeV electrons



Neutron source file case

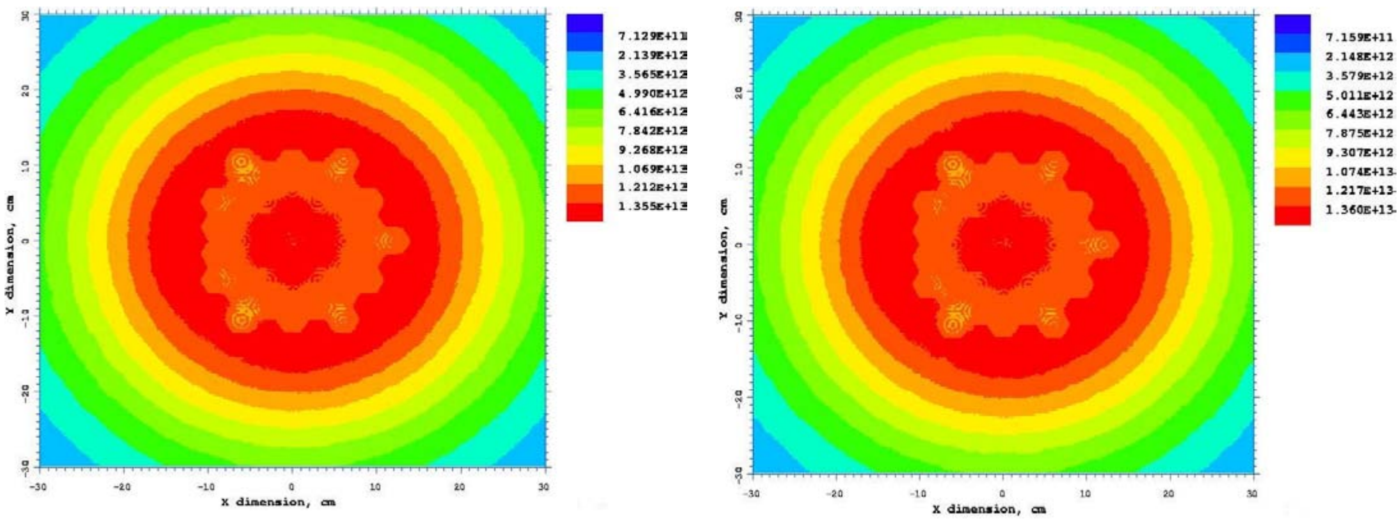
Electron source case

Figure 3. Comparison of R-Z thermal neutron flux (E<0.1 MeV) maps calculated using MCNPX with neutron source file and MCNPX transporting 200 MeV electrons



Neutron source file case Electron source case

Figure 4. Comparison of X-Y fast neutron flux (E>0.1 MeV) maps calculated using MCNPX with neutron source file and MCNPX transporting 200 MeV electrons



Neutron source file case Electron source case

Figure 5. Comparison of R-Z thermal neutron flux (E<0.1 MeV) maps calculated using MCNPX with neutron source file and MCNPX transporting 200 MeV electrons

Table I. Comparison of neutron flux and energy deposition calculated using MCNPX with neutron source file and MCNPX transporting 200 MeV electrons

Calculation Method	Average neutron Flux along the core ($n/cm^2 \cdot s$)	Average neutron flux along the target ($n/cm^2 \cdot s$)	Neutron energy deposition (kW)		
			Target	Core	Reflector
Neutron source file	2.546e+13 (± 1.02 %)	3.142e+13 (± 0.99 %)	2.834 (± 1.00 %)	233.01 (± 1.04 %)	2.38 (± 1.04 %)
electrons Transport	2.559e+13 (± 1.12 %)	3.151e+13 (± 1.12 %)	2.842 (± 1.10 %)	234.41 (± 1.14 %)	2.39 (± 1.14 %)

III. ADS burnup calculations

The results in the previous section show that the generated neutron source file can be used to characterize and define the performance of the electron driven subcritical assembly. Therefore, the MCB computer code was used with the neutron source file to perform burnup calculation utilizing its fixed source burnup capability. MCNPX and MCB burnup calculations using k-code mode were also performed to study the impact on the results relative to the fixed source results from MCB.

III.1 Burnup calculational model

The burnup equations use neutron fluxes, nuclide number densities, and cross sections to determine the time-dependent nuclide inventory [5]. Time evolution of nuclide number densities is calculated using linear transmutation chains that are prepared for every burnup zone and time step [6]. The assembly is divided into depletion zones where the spatial neutron flux distribution and the neutron spectrum do not vary much inside each zone. In the MCB and MCNPX calculational models, the 35 fuel assemblies are divided into 3 burnup zone based on their radial locations, as shown in Figure 6 and each target plate is treated a separate burnup zone. Therefore, the total number of burn zones in the calculational model is 15 zones.

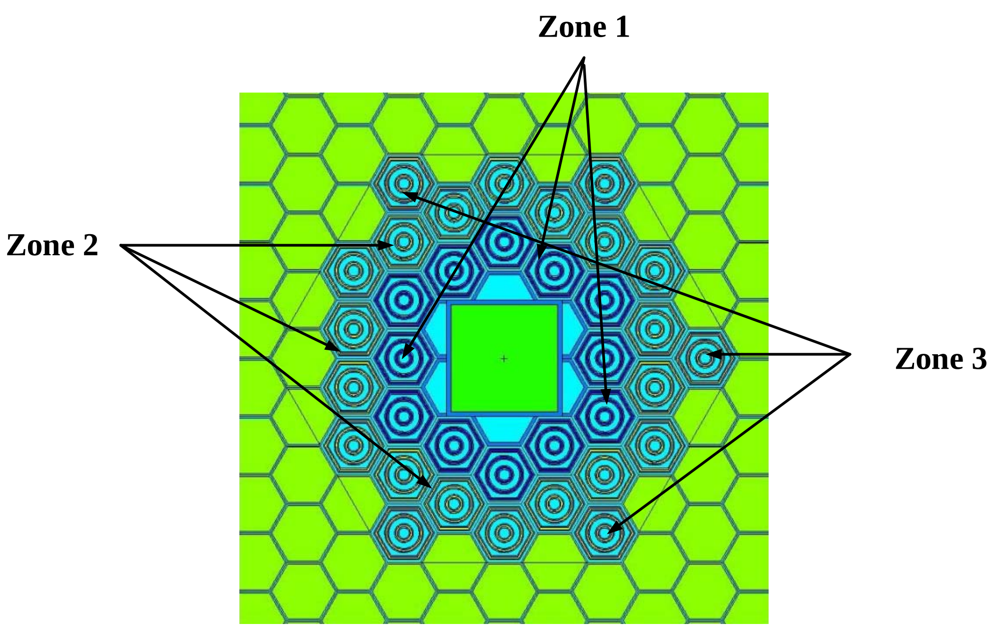


Figure 6. Burnup zones for the ADS fuel assemblies

In the burnup equations, the change in nuclide concentration depends on the time-integrated neutron flux. However, the time-dependent neutron flux depends on the nuclide density during the depletion time. Therefore the burnup equations are non-linear and too complicated to solve analytically. Both MCB and MCNPX assume that the neutron flux is constant throughout the current depletion time step. However the difference between the two computer codes is related to the method used to determine the neutron flux value during the time step. In MCB, the neutron flux is calculated at the beginning of the time step and it is assumed constant through the whole time step. This approximation works well only if the flux shape and spectra change between the two time points is trivial. In MCNPX, a more accurate method called predictor-corrector technique, which is widely used in the current burnup codes is adopted. And it can be summarized into three steps [5]:

1. A burnup calculation is performed for half of the time step using the neutron flux generated at the beginning of the time step. This is known as predictor step.
2. MCNPX is used to calculate new neutron flux and collision densities at the middle of the time step.
3. The recalculated neutron flux and collision densities are used for burnup calculation for the full time step. This step is known as corrector step.

For this predictor-corrector technique, it is assumed that the neutron flux and collision densities recalculated at the middle of time step are the average values for the entire time step. This method assumes that the flux shape and spectra change within the time step is linear, therefore it is better than the simple method used in MCB, and it is widely used in the burnup calculations. This predictor-corrector technique requires fewer depletion steps because a longer depletion time step can be used. However, two transport calculations are required for each time step in MCNPX instead of only one for MCB calculation. Even with the predictor-corrector technique, burnup calculation with large time steps might cause large flux shape and spectra changes during the time steps, which may lead to inaccurate results. Therefore, small time steps should be still used to calculate the average values of the neutron flux and spectrum accurately for each time step.

III.2 Burnup results from fixed source and fission source calculations

As explained previously, small time steps should be used to avoid large variations in the neutron flux distribution, neutron spectrum, and nuclide densities of each burnable zone during a single time step. For the ADS under consideration, the power and neutron flux values are relatively low, which allows for large time steps relative to conventional nuclear power reactors. In this calculation, the time step size is 45 days except at the beginning of the burn cycle. Small time steps are used to show the fast change in the system reactivity and neutron flux due to the buildup of Xe and Sm fission products. The calculated k-eff history is shown in Figure 7 assuming the facility is operated at full power continuously for ~1 year.

The MCB – source mode results shown in Figure 7 are obtained with MCB calculations using the neutron source file, which was generated by MCNPX calculation starting with electron beam. At each burnup step, an MCB k-code mode calculation was performed to get k-eff of the assembly. In this calculation neutron flux and reaction rates were not calculated. After the MCB k-code calculation, another MCB calculation with the neutron source file was performed to calculate the neutron flux and the reaction rates, which were used for the burnup calculation.

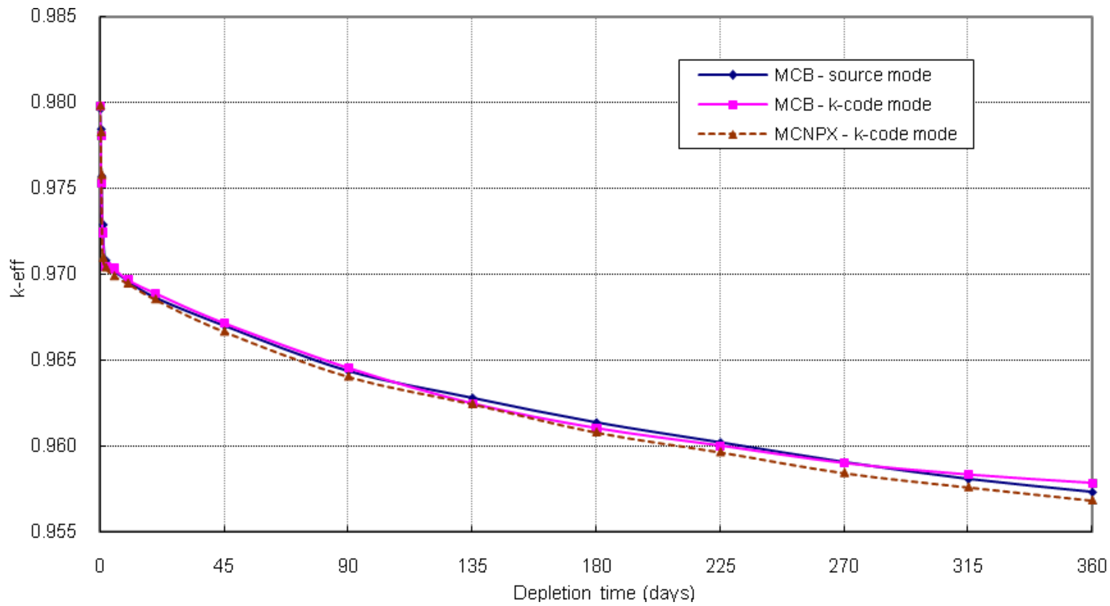


Figure 7. k-eff history calculated by MCB and MCNPX

The neutron source strength is constant during all the time steps, while the neutron flux and fission power of the subcritical assembly decrease due to the consumption of the fissile material as shown in Figure 8. The fission power of the subcritical assembly was calculated by MCB at each burnup step.

Burnup calculations were also performed by MCB and MCNPX in k-code mode, using the fission power shown in Figure 8 to normalize the neutron flux at each burnup time step. The corresponding k-eff results are also shown in Figure 7. The results of Figure 7 show the difference in k-eff values calculated with fixed source mode and k-code mode using fission source with adjusted fission power are within the statistical error of the Monte Carlo calculations. This conclusion is valid for the subcritical assembly under consideration with low power density. The MCB and MCNPX calculations used nuclear data libraries based on ENDF-B/VI.

The nuclei number densities of some important fission products and actinides of the three zones of the critical assembly fuel are obtained from MCB burnup calculation using the source mode and k-mode. The obtained results are shown in Figures 9 through 12. From the results, it can be seen that using k-mode instead of source mode in MCB burnup calculation does not cause apparent error for the number density of fission products and actinides. MCNPX has the advantages of updated burnup physics relative to MCB and the capability of parallel computation techniques. Using MCNPX k-code mode cannot calculate the power change during the burnup time. To get the power level at each burnup time step, additional MCNPX source calculation using electrons and updated nuclei number densities is needed to calculate the assembly power. This process is time consuming and it increases the computation burden considerably. Therefore MCB is still utilized for the ADS burnup analysis.

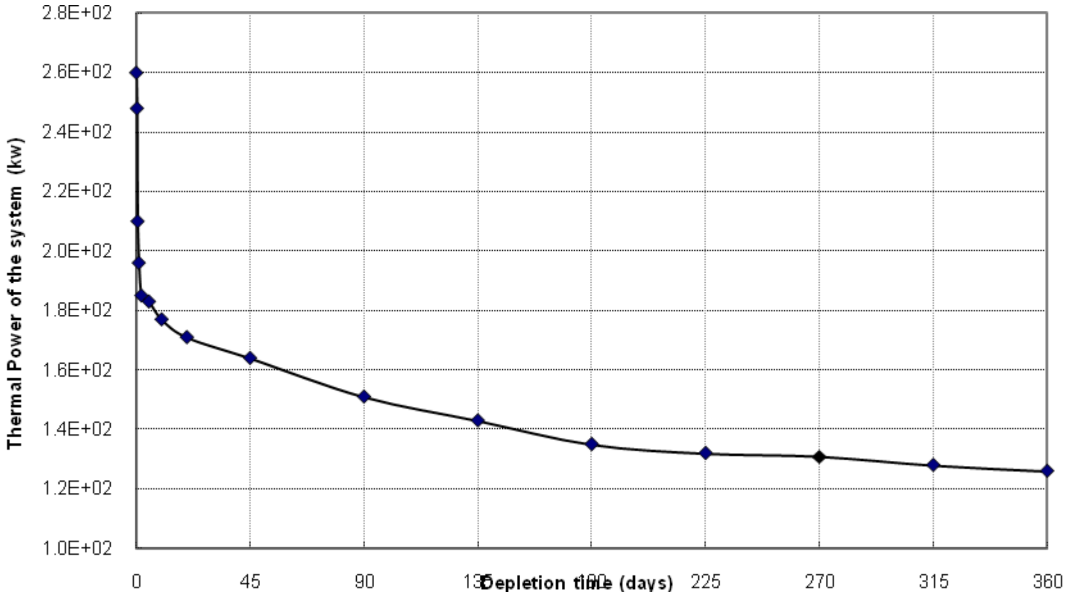


Figure 8. Fission power change during the burnup cycle

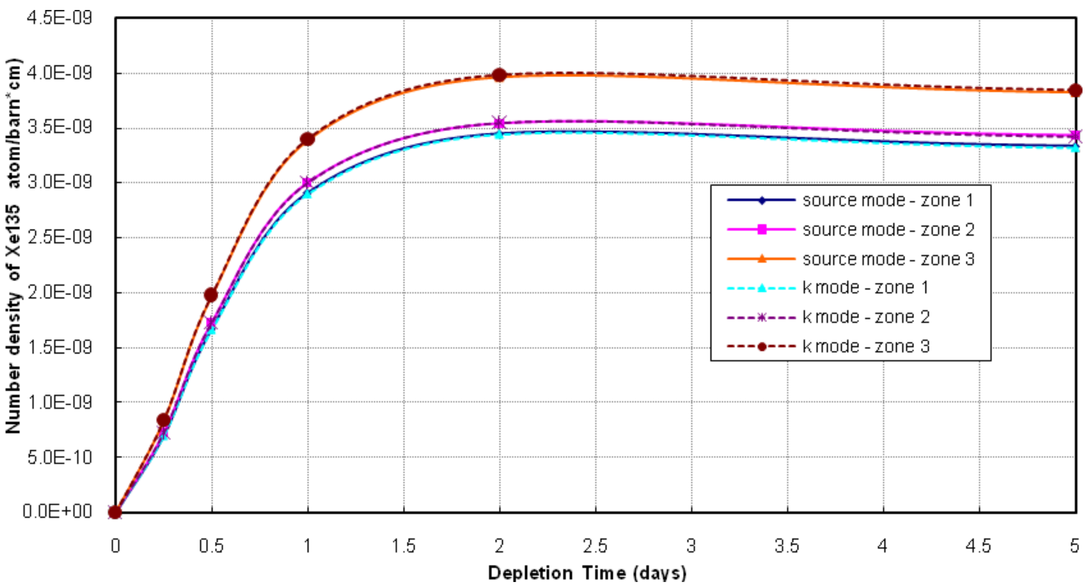


Figure 9. Xe-135 nuclei number density during the first 5 days

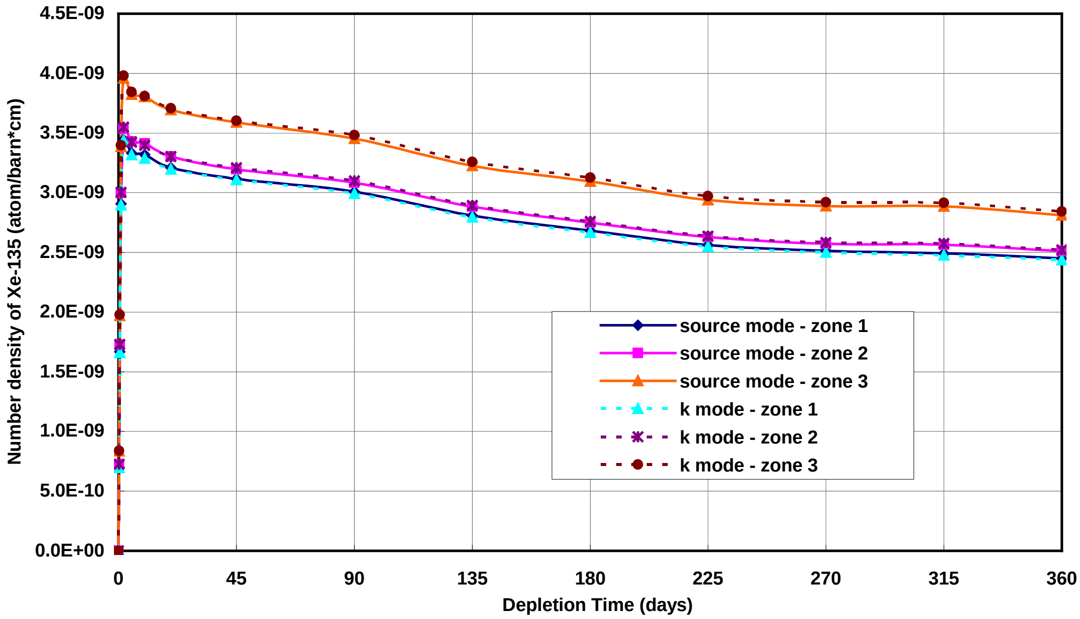


Figure 10. Xe-135 nuclei number density during the first 360 days

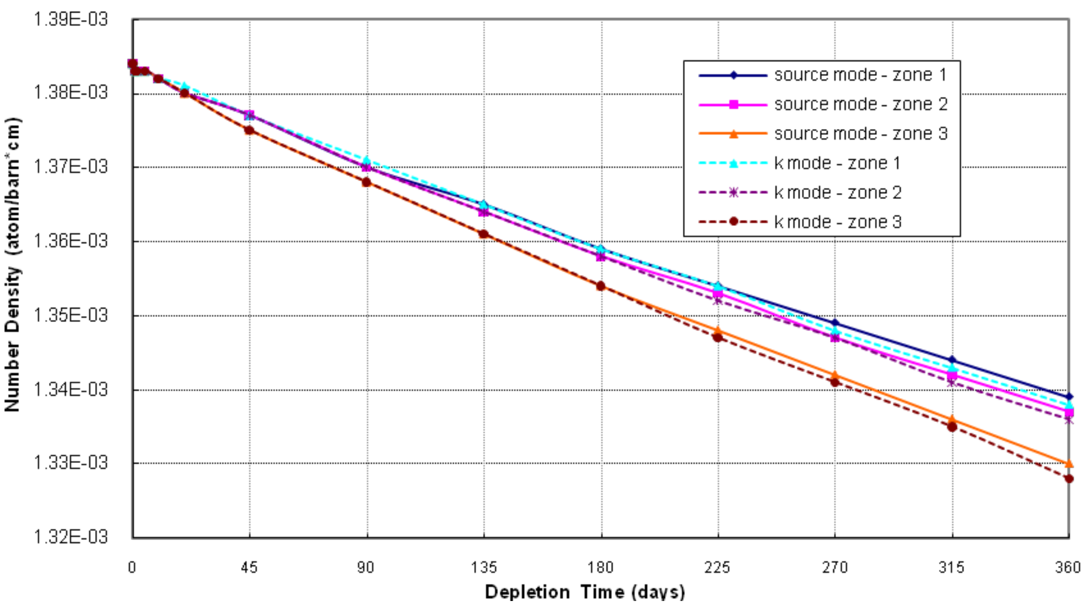


Figure 11. U-235 nuclei number density during the first 360 days

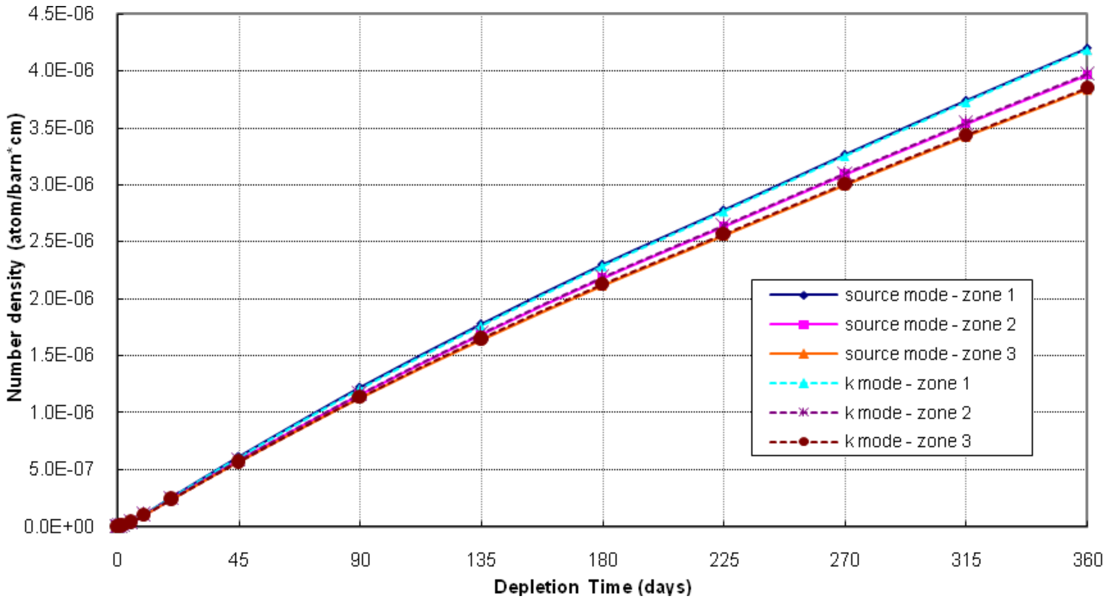


Figure 12. Pu-239 nuclei number density during the first 360 days

IV. Burnup history with the addition of fresh fuel assemblies

During the operation, the subcritical assembly reactivity decreases due to the fissile material consumption and the buildup of fission products. Unlike fission power reactors, which can regain reactivity through the use of burnable poison or withdrawing control rods, the accelerator driven systems do not have such possibilities. The subcritical assembly is driven by the electron beam, which has a fixed 100 KW Power. Therefore during the operation, the neutron flux and the power level of the subcritical assembly decrease, which impact the facility performance. To regain the reactivity as well as the neutron performance of the facility, fresh fuel assemblies are added during the operation. The burnup and the refueling scheme are studied in this section.

The burnup history of the subcritical assembly is divided into stages. After each stage, the facility is stopped for several days and one fresh fuel assembly is added. Therefore, the gained reactivity due to the addition of the fresh fuel assembly should compensate that lost reactivity during the last burnup stage. The burn stages should be carefully determined to insure that the added reactivity is equal to the lost reactivity for maintaining the desired subcriticality level. The burnup analyses should accurately predict the reactivity change during operation. As discussed in the previous section, Monte Carlo depletion code MCB is utilized for this calculation with the banked neutron source file generated by MCNPX to preserves the neutron yield from photonuclear reactions generated by the electron beam.

At the beginning, the subcritical assembly is loaded with 35 fresh fuel assemblies, as shown in Figure 6. The k-eff is 0.98062 and the power and flux maps generated from the use of the 100 KW electron beam with 200 MeV electrons are calculated by MCNPX and shown in figures 13 through 15. The high power density in the target region is caused by the electron energy deposition. The power density distribution in the fuel region is relatively uniform.

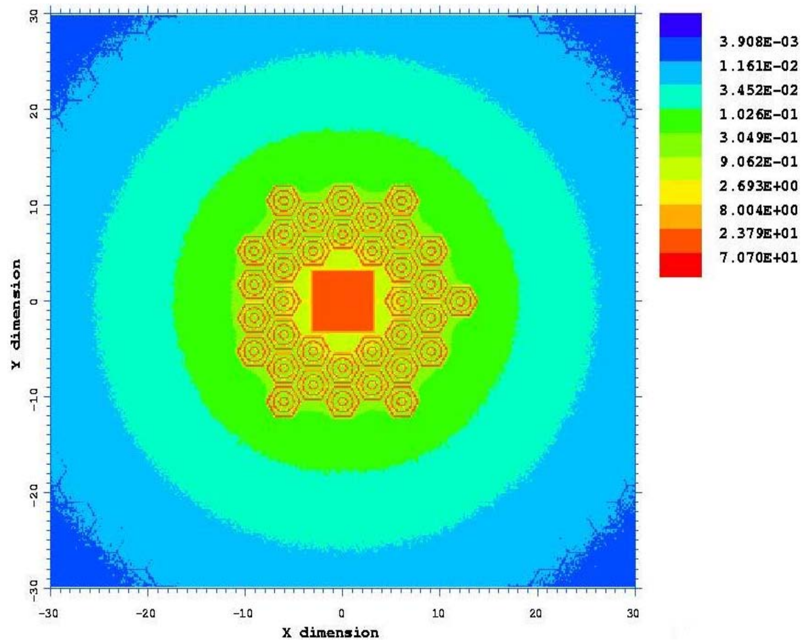


Figure 13. Energy deposition map of the subcritical assembly with 35 fresh fuel assemblies, W/cm^3

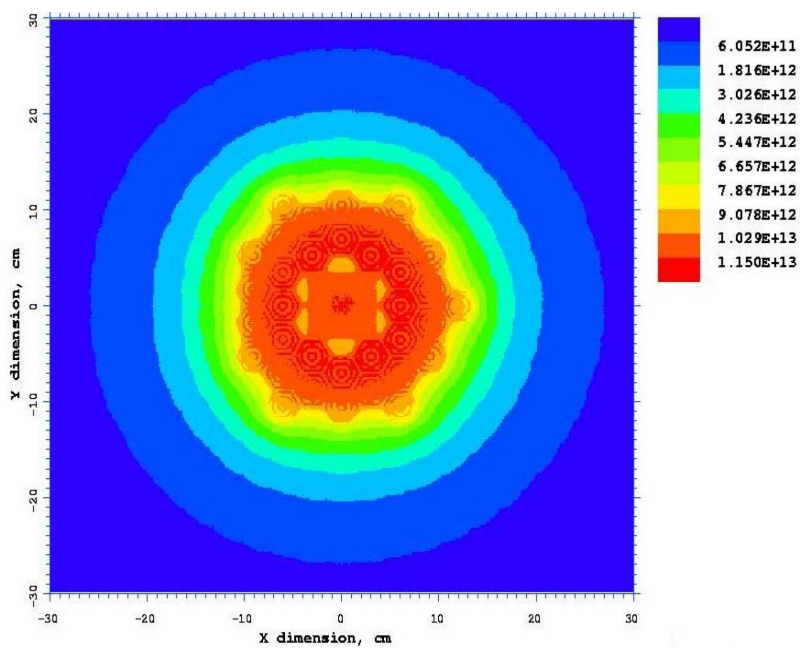


Figure 14. Fast neutron flux (E > 0.1 MeV) map of the subcritical assembly with 35 fresh fuel assemblies, n/cm^2.s

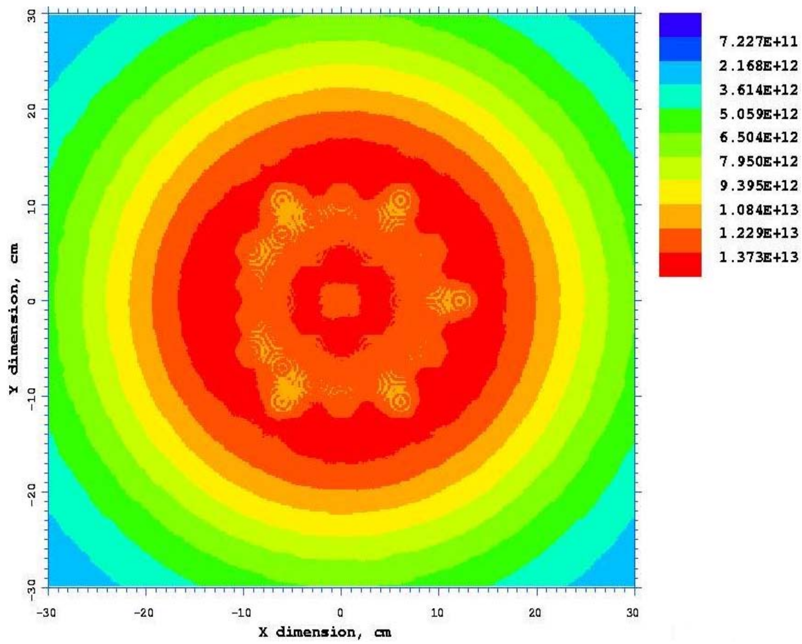


Figure 15. Thermal neutron flux (E < 0.1 MeV) map of the subcritical assembly with 35 fresh fuel assemblies, n/cm^2.s

IV.1 First Burnup Stage

The first burnup stage is 90 days. The reactivity of the subcritical assembly during this burn stage and the reactivity change during the shutdown time after the 90 days of operation are shown in Figures 16 and 17, respectively. The burnup cycle is designed so the subcritical assembly reactivity can get back to the original value by adding one fresh fuel assembly as shown in Figure 18.

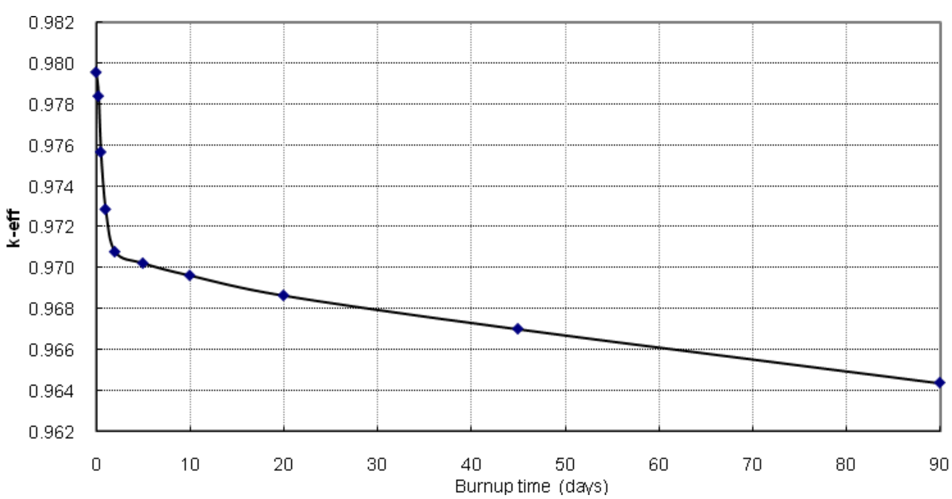


Figure 16. k-eff as a function of time during the first burnup stage

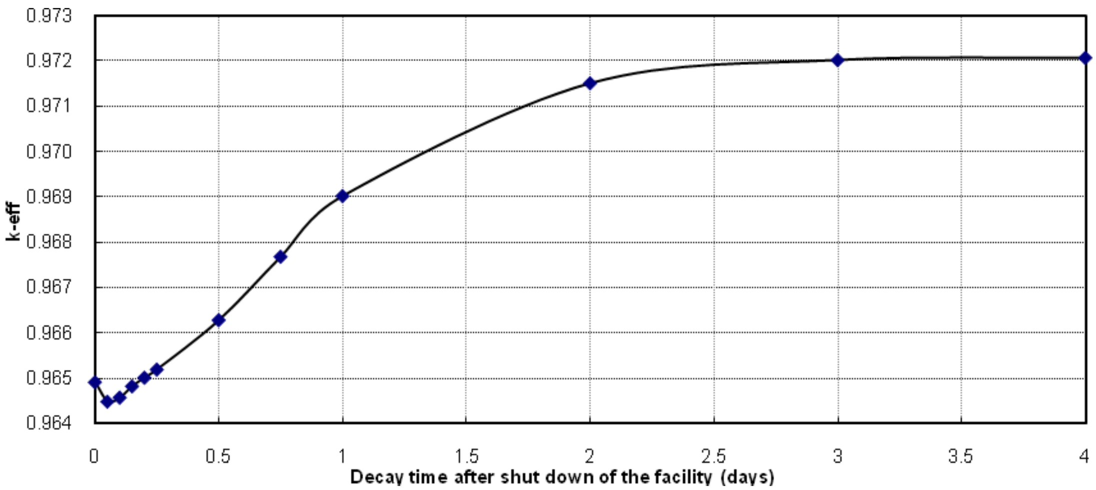
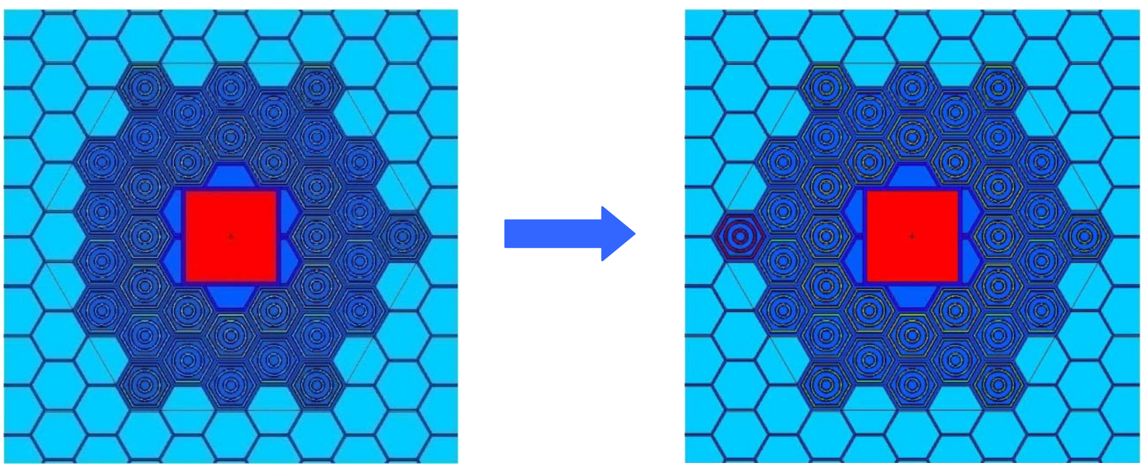


Figure 17. k-eff as a function time during the shutdown time after the first burnup stage



35 fuel assemblies 36 fuel assemblies

Figure 18. Subcritical assembly configuration with one fresh fuel assembly after the first burnup stage

After the first burn stage, the K-eff of the subcritical assembly decreases by ~ 1500 pcm as shown in Figure 16. During the shutdown after the first burnup stage, Xe-135 decays and K-eff value of the subcritical assembly increases by ~ 700 pcm as shown in Figure 17. Adding one fresh fuel assembly after 3 shutdown days increases k-eff to 0.97936 (±35 pcm), which is appropriate value to start the second burn stage. The neutron flux and power maps of the subcritical assembly with 36 fuel assemblies at the beginning of the second burnup stage are calculated by MCNPX using the nuclei number densities from MCB burnup results. The obtained results are shown in Figure

19 through 21. The neutron flux and the power density distribution do not show any peaking. Therefore introducing one fresh fuel assembly into the system doesn't cause any thermal hydraulics concerns.

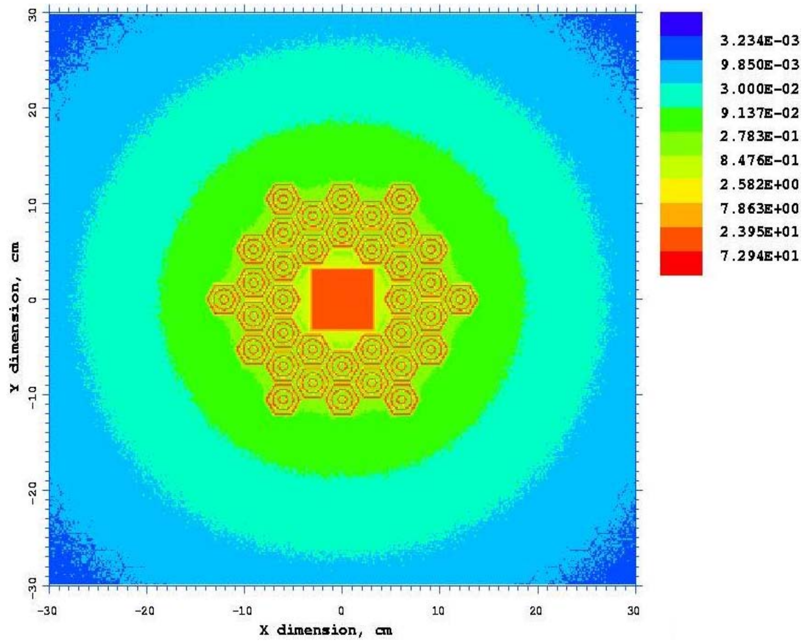


Figure 19. Energy deposition map of the subcritical assembly with 36 fuel assemblies, W/cm^3

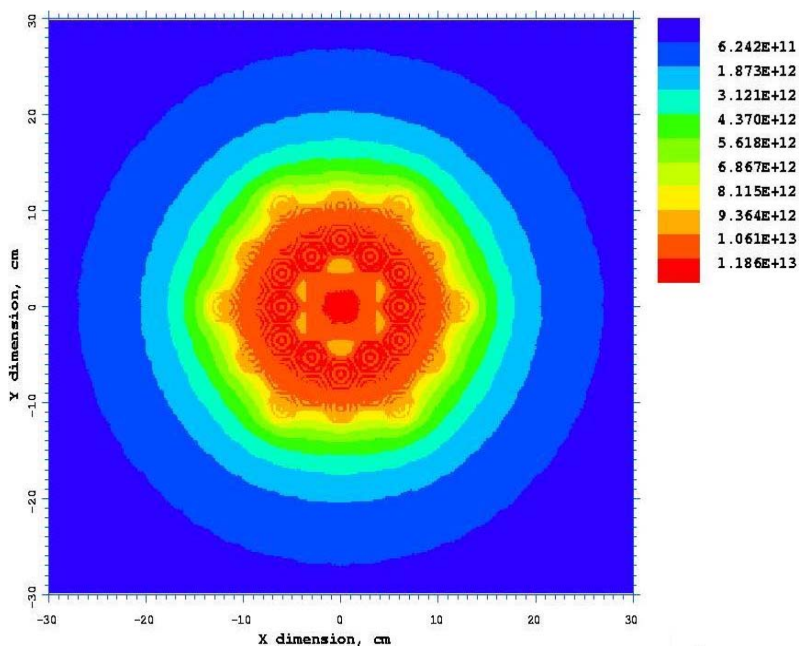


Figure 20. Fast neutron flux (E > 0.1 MeV) map of the subcritical assembly with 36 fuel assemblies, $n/cm^2.s$

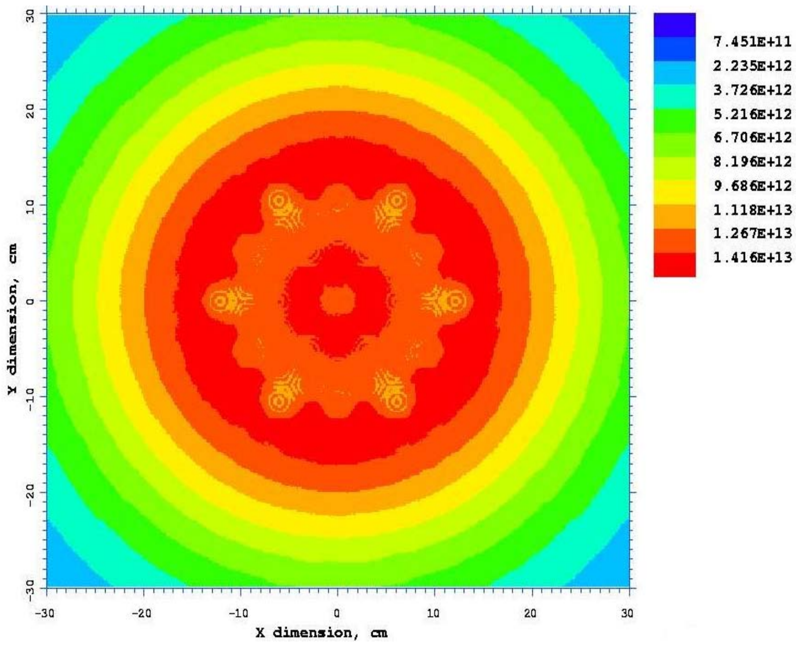


Figure 21. Thermal neutron flux (E < 0.1 MeV) map of the subcritical assembly with 36 fuel assemblies, n/cm^2.s

IV.2 Second burnup stage

In the first burn stage, the decay of the fission products after shutting down the facility increases k-eff by ~ 700 pcm and the addition of one fresh fuel assembly results in another increase of ~700 pcm. The burn cycle of 90 days reduces the k-eff of the subcritical assembly by ~ 1400 pcm, which explains the 90 days selected for the first burn stage. The second burnup stage has also 90 days. The k-eff of the subcritical assembly as a function of time during and after the second burn stage is shown in Figures 22 and 23. At the end of the second burnup stage, the k-eff of the subcritical assembly is ~ 0.966. After 3 days of shut down, the k-eff increase to 0.97438. Adding one fresh fuel as shown in Figure 24 increases to K-eff to 0.98119 (±29 pcm).

The neutron flux and power map of the subcritical assembly with 36 fuel assemblies at the beginning of the third burnup stage are calculated with MCNPX using the nuclei number density from MCB burnup results. The obtained maps are shown in Figures 25 through 27.

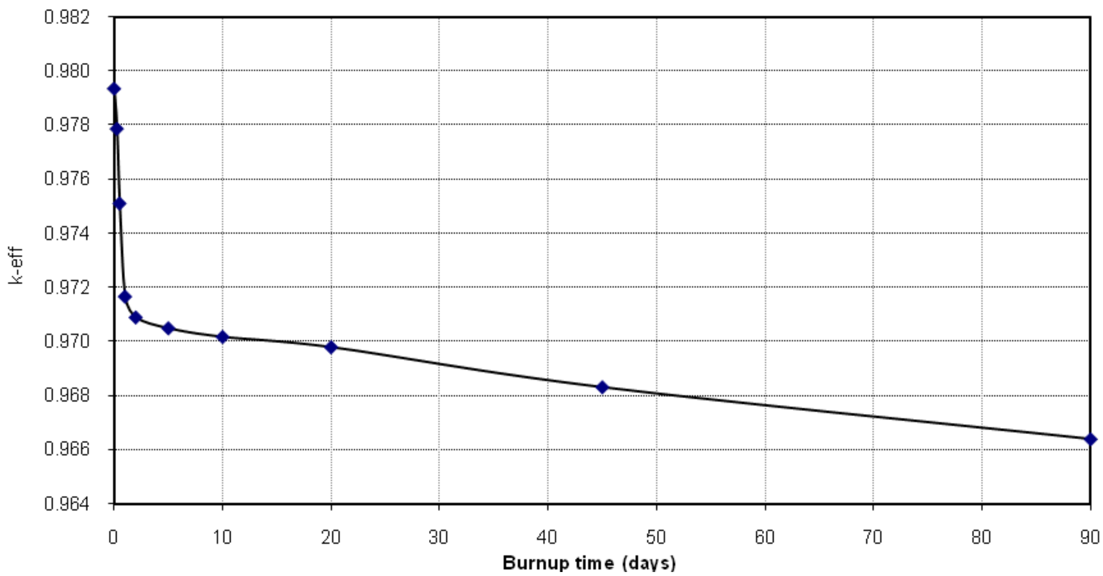


Figure 22. k-eff as a function of time during the second burnup stage

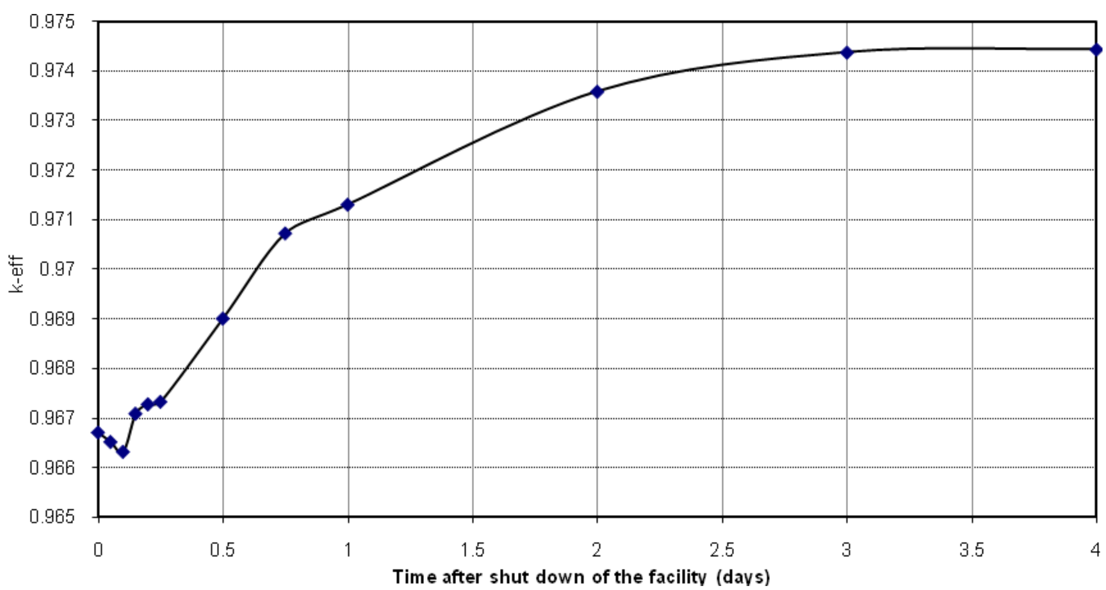
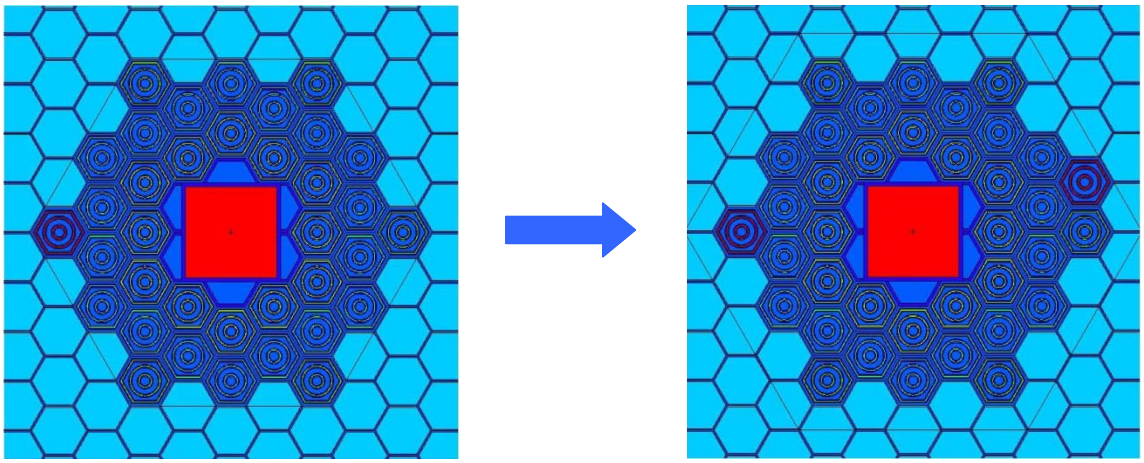


Figure 23. k-eff as a function time during the shutdown time after the second burnup stage



36 fuel assemblies 37 fuel assemblies

Figure 24. Subcritical assembly configuration with one fresh fuel assembly after the second burnup stage

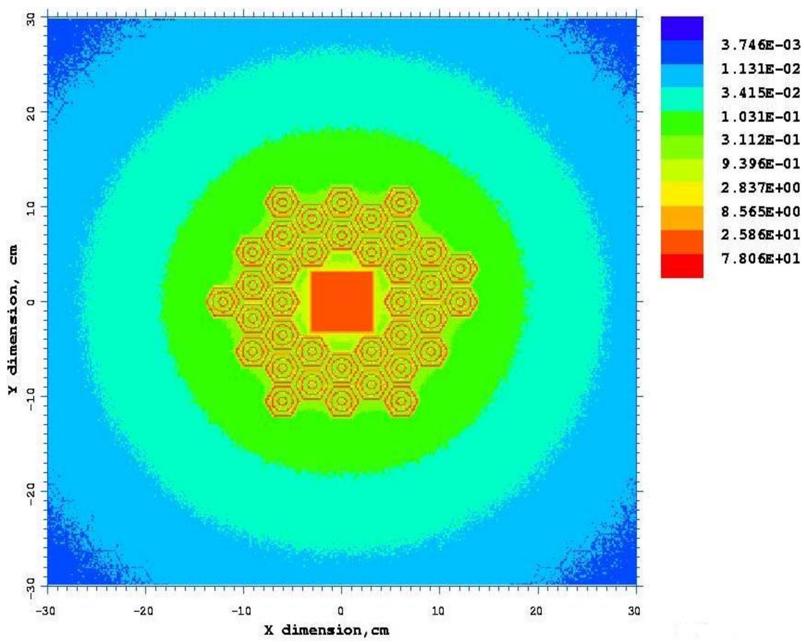


Figure 25 Energy deposition map of the subcritical assembly with 37 fuel assemblies, W/cm^3

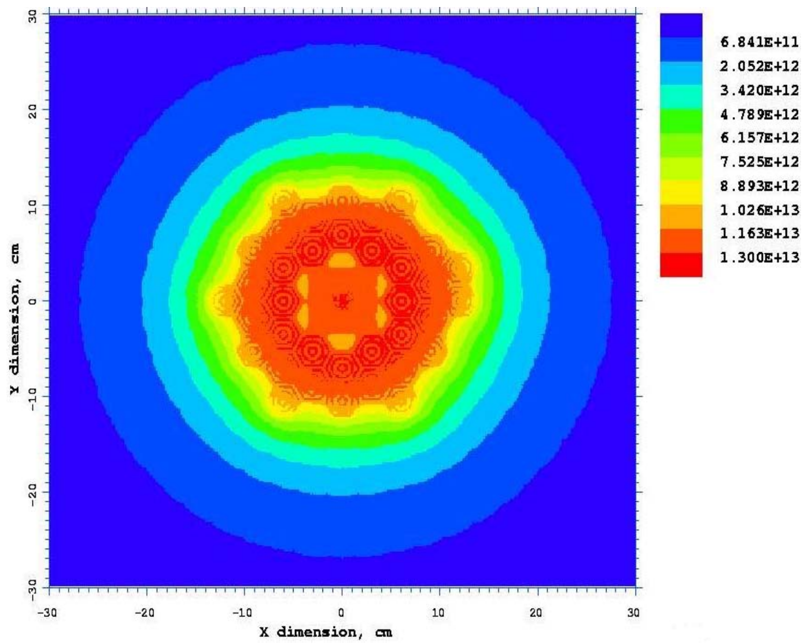


Figure 26. Fast neutron flux (E > 0.1 MeV) map of the subcritical assembly with 37 fuel assemblies, n/cm^2.s

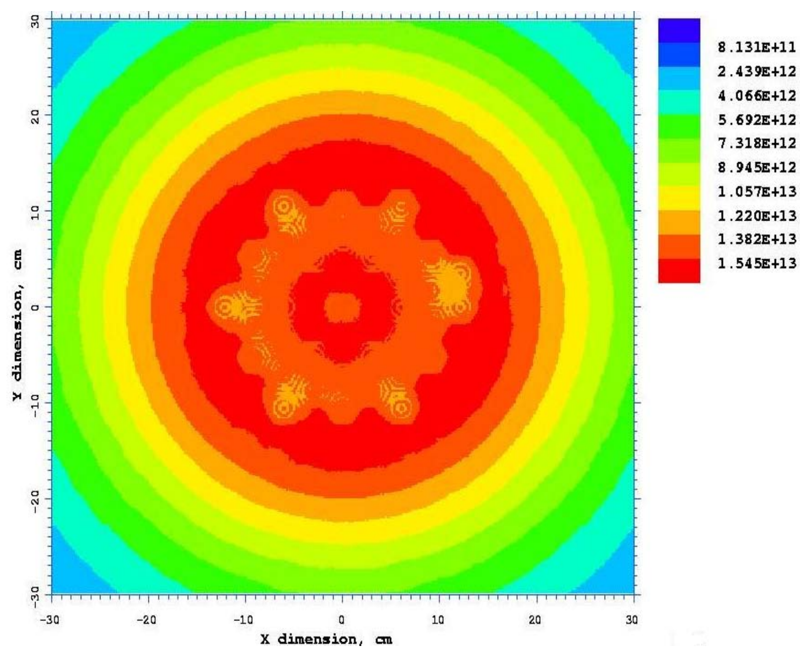


Figure 27. Thermal neutron flux (E < 0.1 MeV) map of the subcritical assembly with 37 fuel assemblies, n/cm^2.s

IV.3 Depletion and fueling stage 3

As explained above, the burnup stage length is selected so the decrease in the k-eff value is 1400 pcm at the end of the burn stage. For the third burnup stage 3 with totally 37 fuel assemblies, the burn time is 225 days because of the slower reactivity change. The k-eff of the subcritical assembly as a function of time during and after the third burn stage is shown in Figures 28 and 29. The neutron flux and power map of the subcritical assembly with 38 fuel assemblies shown in Figure 30 at the beginning of the fourth burnup stage are calculated with MCNPX using the nuclei number density from MCB burnup results. The obtained maps are shown in Figures 31 through 33.

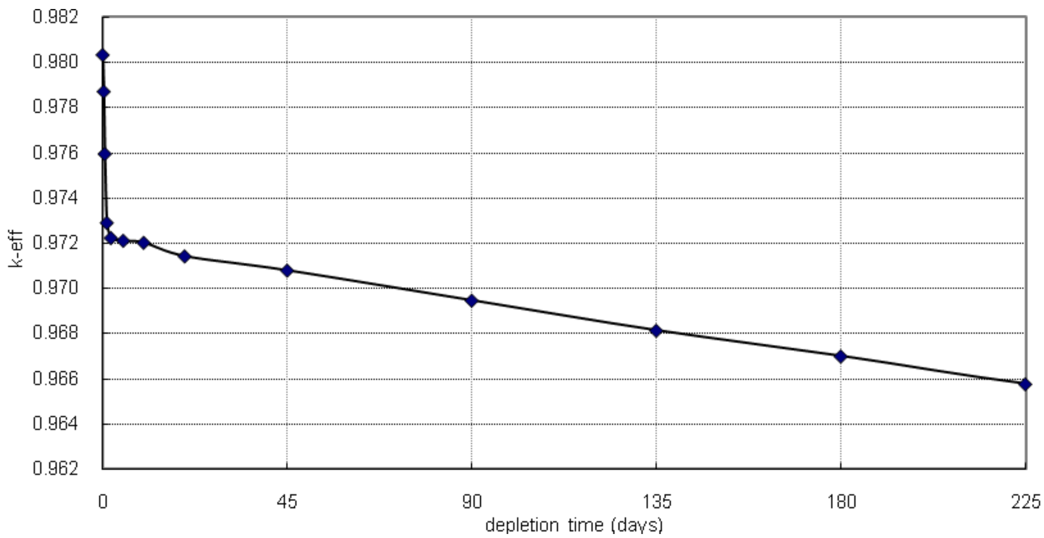


Figure 28 k-eff as a function of time during the third burnup stage

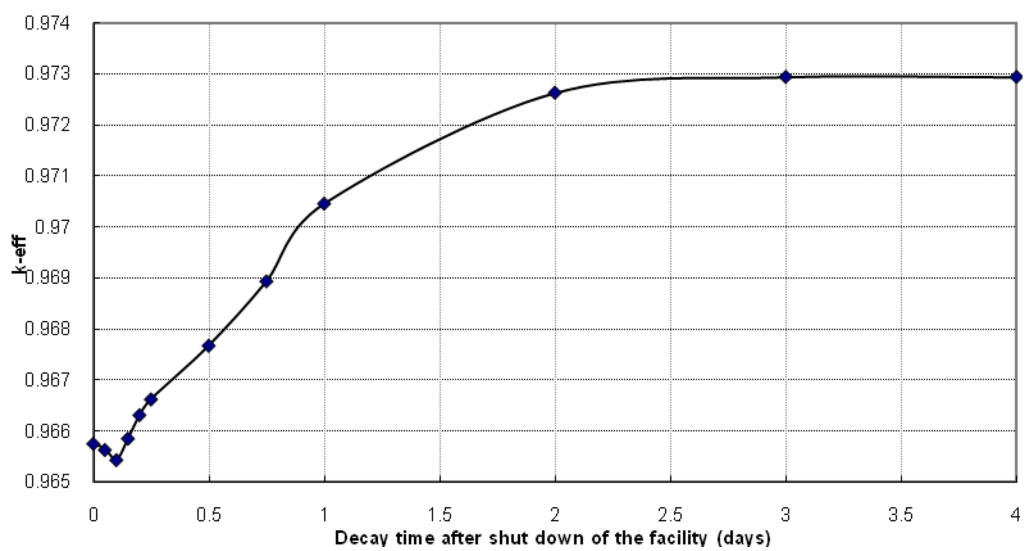
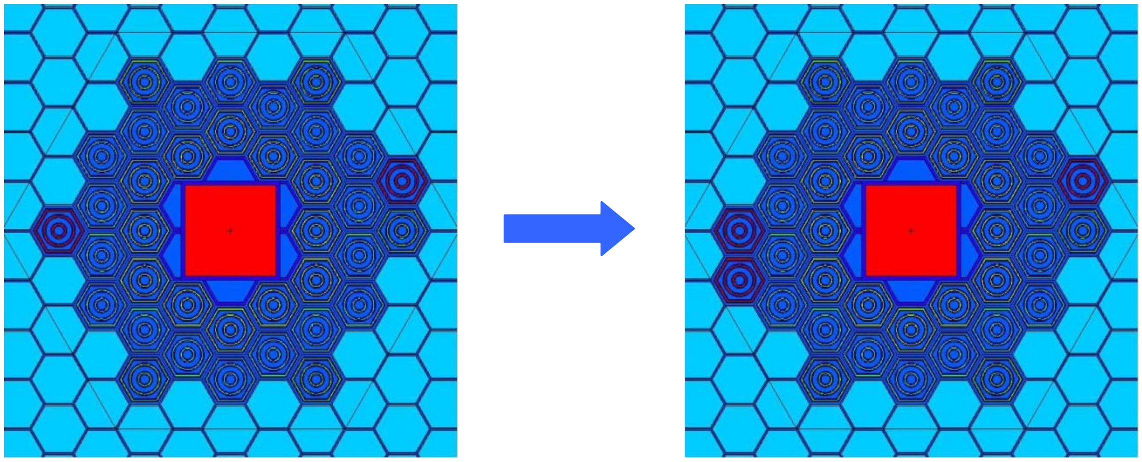


Figure 29 k-eff as a function time during the shutdown time after the third burnup stage



37 fuel assemblies

38 fuel assemblies

Figure 30. Subcritical assembly configuration with one fresh fuel assembly after the third burnup stage

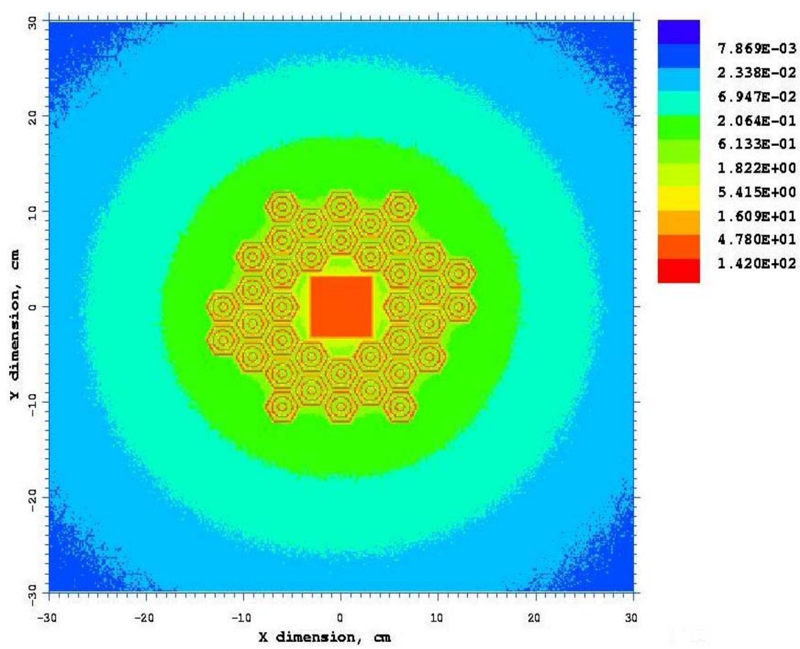


Figure 31. Energy deposition map of the subcritical assembly with 38 fuel assemblies, W/cm^3

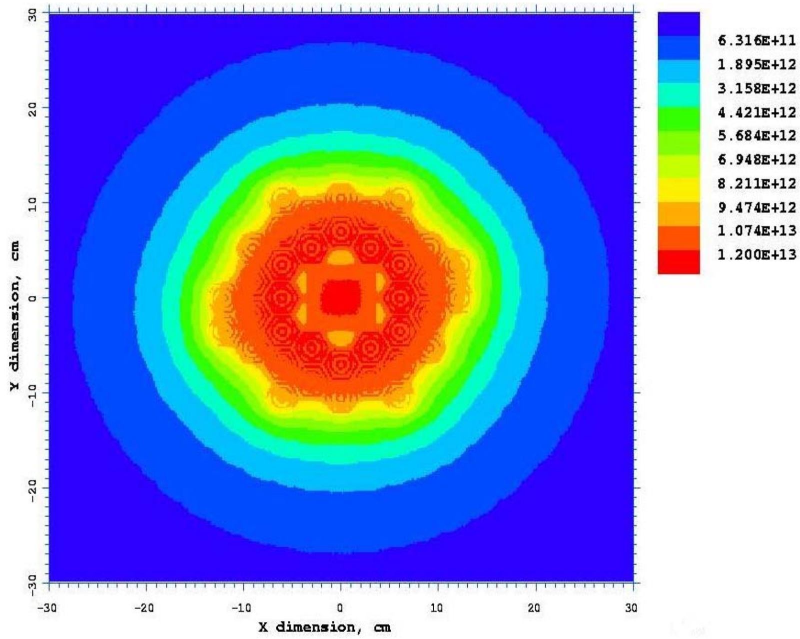


Figure 32. Fast neutron flux (E > 0.1 MeV) map of the subcritical assembly with 38 fuel assemblies, n/cm^2.s

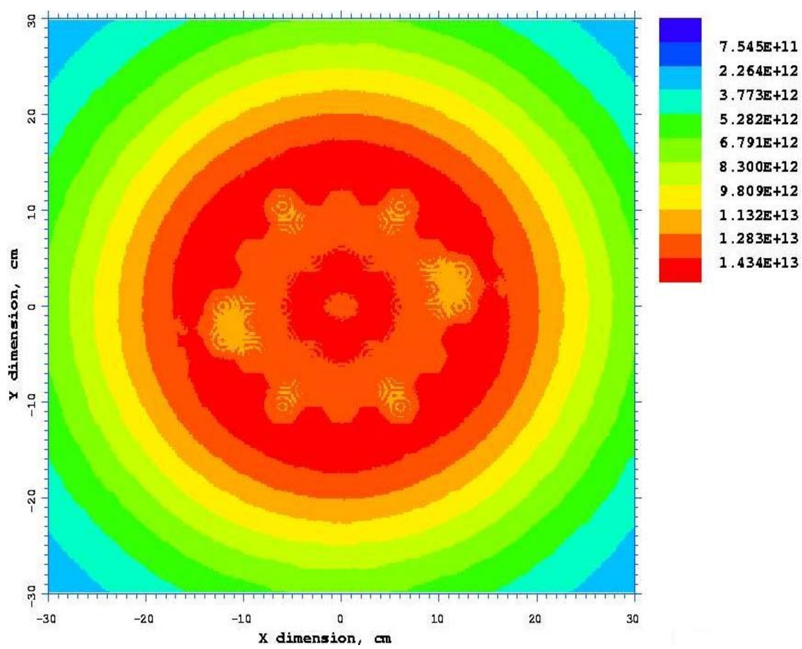


Figure 33. Thermal neutron flux (E < 0.1 MeV) map of the subcritical assembly with 38 fuel assemblies, n/cm^2.s

IV.4 Subcritical assembly performance parameters during the different burnup stages

In the previous sections, the fuel cycle to operate the subcritical assembly was studied for different burnup stages to determine the fueling strategy to maintain the subcriticality level close to the designed value. The important parameters from this study, which are calculated by MCB and MCNPX are summarized. The k-eff value at different time points, including BOC (Begin of Cycle), EOC (End of Cycle) and after three days of shut down from MCB calculation are shown in Table II. The performance parameters at the beginning of each depletion stage are calculated by MCNPX and given in Table III.

Table II. K-eff values calculated by MCB during the different burnup stages

Burnup Stage	Number of fuel assemblies	Burnup time (days)	k-eff		
			BOC	EOC	after 3 days of shut down
1	35	90	0.97950 (± 33 pcm)	0.96437 (±38 pcm)	0.97202 (±34 pcm)
2	36	90	0.97936 (±35 pcm)	0.96641 (±30 pcm)	0.97438 (±33 pcm)
3	37	225	0.98033 (±24 pcm)	0.96576 (±22 pcm)	0.97294 (±25 pcm)

Table III. Performance parameters of the subcritical assembly during the different burnup stages

Burnup Stage	Number of fuel assemblies	k-eff	Neutron flux along the core (n/cm^2·s)	Neutron flux along the target (n/cm^2·s)	Fission heat (kW)*	
					Target assembly (KW)	Fuel assemblies (KW)
1	35	0.98062 (±12 pcm)	2.543e+13 (±0.64%)	3.134e+13 (±0.62%)	5.98 (±0.63%)	245.7 (±0.65%)
2	36	0.97999 (±12 pcm)	2.636e+13 (±0.60%)	3.255e+13 (±0.58%)	6.25 (±0.59%)	258.9 (±0.61%)
3	37	0.98173 (±12 pcm)	2.873e+13 (±0.64%)	3.537e+13 (±0.62%)	6.85 (±0.63%)	287.7 (±0.65%)
4	38	0.98026 (±12 pcm)	2.677e+13 (±0.62%)	3.304e+13 (±0.60%)	6.50 (±0.61%)	271.06 (±0.63%)

* the fission energy doesn’t account for the energy of the neutrinos and the fission product decay

The results of Table III show that the neutron flux at the beginning of each burnup stage is preserved. This indicates that the developed fueling scheme works satisfactory. At BOC of each depletion stage, the k-eff calculated by MCNPX is consistently higher than that calculated by MCB, although both codes use ENDF-B/VI. This small difference is due to the different modes of ENDF/B-VI data libraries used by the two computer codes and the statistical error of the calculations. In addition, the MCB calculation tracks less number of particles per run to reduce the computational time due to the large number of time steps. Furthermore, the parallel technique adopted in MCB, PVM (Parallel Virtual Machine), loses its efficiency as the number of processors increase above six, which limits the practicality of performing large number of particles per run and the long burnup calculations using MCB.

V. Summary and Conclusions

Monte Carlo codes MCB and MCNPX have been utilized successfully for performing burnup analyses for KIPT accelerator driven subcritical system. A hybrid process that couples MCNPX and MCB, has been developed, which exploits both the electrons transport capability of MCNPX and the depletion capability of MCB for subcritical system. In this process, a neutron source file is generated from MCNPX calculation, which preserves the neutrons yield from photonuclear reactions and their characteristics. This file is utilized by MCB as an external neutron source for the subcritical assembly. Using this hybrid process, the fuel cycle was studied. The neutron performance of the KIPT facility is preserved with adding one fresh fuel assembly into the system after each burnup stage while maintaining the peak value of k-eff at 0.98, which satisfies the safety requirement. In addition, it was observed that performing the burnup calculations using k-mode provides correct results as long as the flux values are adjusted each burn stage to match the subcritical assembly power. However, the power values needs to be obtained using separate set of calculations, which complicate the analyses. The low power density of the subcritical assembly and neutron spectrum of the neutron source are the reasons for this conclusion. The neutron spectrum of the source file is very close to the fission spectrum from the fuel point of view.

Reference

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. Y. Gohar, I. Bolshinsky, J. Bailey, H. Belch, D. Naberezhnev and J. Duo, “Accelerator Driven Sub-Critical Facility: Conceptual Design Development”, NIMPR A562, Proc. of AccApp05, 23 June 2006, Pages 870-874.

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^{TM} USER’S MANUAL”, Los Alamos Report, LA-CP-05-0369, April 2005.

5. John S. Hendricks, Gregg W. Mckinney, Michael L. Fensin, “MCNPX, version 26e”, Los Alamos Report, LA-UR-07-6632, Novemberl 2007.

6. Jerzy Cetnar, W. Gudowski, J. Wallenius, “ User Manual for Monte Carlo Continuous Energy Burnup (MCB) Code – Version 1C”, NEA-1643/01, August, 2002

7. J. F. Briesmeister, Editor: “MCNP –A General Monte Carlo Code N-Particle Transport Code”, LANL report LA-12625-M, Version 4B (March 1997)

8. J. Cetnar: "A Method of Transmutation Trajectories Analysis in Accelerator Driven System", Proc. IAEA Technical Committee Meeting on Feasibility and Motivation for Hybrid Concepts for Nuclear Energy Generation and Transmutation, Madrid, 17-19 Sept. 1997 (in pub.)

9. 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.

10. X-5 Monte Carlo Team, “MCNP – A General Monte Carlo N-particle Transport Code, Version 5, Volume II User’s Guide”, Los Alamos Report, LA-UR-03-0245, October 2005.

11. W. P. Swanson, “Radiological Safety Aspects of the Operation of Electron Linear Accelerators”, IAEA, Vienna, 1979, Pages 61~94.

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