Utilizing the burnup capability in MCNPX to perform depletion analysis of an MNSR fuel

Emmanuel Boafo[1]

[1]*Ghana atomic Energy Commission, Ghana*

E-mail: kwesiboafo@yahoo.com

The burnup capability in the MCNPX code was utilized to perform fuel depletion analysis of the MNSR LEU core by estimating the amount of fissile material (U-235) consumed as well as the amount of plutonium formed after the reactor core expected life. The decay heat removal rate for the MNSR after reactor shut down was also investigated due to its significance to reactor safety. The results show that 0.568 % of U-235 was burnt up after 200 days of reactor operation while the amount of plutonium formed was not significant. The study also found that the decay heat decreased exponentially after reactor shutdown confirming that the decay heat will be removed from the system by natural circulation after shut down and hence safety of the reactor is assured.