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# Burnup-dependent core neutronics analysis and calculation of actinide and fission product inventories in discharged fuel of a material test research reactor

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**Abstract**

The System for Analysis of Reactor Core (SARC) has been developed for burnup analysis of nuclear reactors using whole core modeling and simulation by coupling the WIMS-D4S and CITATION. The system has the capability to use 1981, 1986 and other WIMS-D libraries recently released by International Atomic Energy Agency, based on the latest cross section data. The SARC was used for the analysis of the first equilibrium core of Pakistan Research Reactor-1 using newly released JENDL3.2 based WIMS-D library. The calculated cycle length and other core related parameters were found in good agreement with the experimental results. Changes in several core parameters such as core reactivity, flux, element-wise power densities, depletion of $^{235}U$ and production of $^{239}Pu$ were also analysed during the cycle. It was found that all these parameters vary linearly with burnup of the core. Moreover, the actinide and fission product inventories in the discharged fuel were also computed.
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*Keywords:* Research reactors; Equilibrium cycle; WIMSD; CITATION; JENDL-3.2; Linear reactivity model

## 1. Introduction

Recently, there has been an increasing interest in performing burnup-dependent core analysis including prediction of the concentration of nuclides in the spent fuel (Murphy and Primm, 2002; Suyama et al., 2002). Spent fuel storage and transportation is a significant problem in many countries which increases the importance of theoretical estimation of burnup credits of actinides and fission products present in the spent fuel (IAEA, 2003). For these reasons a burnup-dependent core neutronics calculation system has been developed named SARC (System for Analysis of Reactor Core). The system was developed in C language for core neutronics analysis of a nuclear reactor, including isotope burnup credit calculations by coupling WIMS-D4S and CITATION (Fowler and Vondy, 1972). SARC calls the codes in their executable forms without any changes to the original validated versions using system function of C language. The development of the system was carried out in two steps. First, WIMS-D4 was upgraded to use updated cross section data and then it was coupled with CITATION for full core depletion analysis.

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The WIMS-D4 code is a general purpose reactor physics lattice cell simulation code (Askew et al., 1966), and is widely used worldwide for thermal research reactor and power reactor calculations. It can solve the energy dependent transport equation in varying degrees of approximations by different methods using one dimensional modeling. It also has the capability to generate burnup and power dependent macroscopic group constants using its multigroup microscopic cross section library. The multigroup cross section library has cross sections and resonance parameters that are independent of both geometry and composition (Graves, 1979). So this one set of microscopic multigroup constants can be applied to all types of thermal reactors. The library associated with the WIMS-D4 package is the United Kingdom based 69-group ‘‘1981’’ WIMS library generated from very old evaluated nuclear data including that from the early 1960s (Taubman, 1975; Aldama et al., 2003). The later versions of the code are distributed through OECD/NEA Data Bank, which include the WIMS-D5B version along with ‘‘1986’’ WIMS library (Halsall, 1991). Since 1986, the WIMS nuclear data libraries have been extensively updated, but the improved versions are only available commercially. The WIMS-D Library Update Project (WLUP) was carried out by the International Atomic Energy Agency to generate updated multigroup cross section libraries for the code to enable scientists and reactor designers to make use of the most recent evaluated nuclear data from different libraries including BROND (Manokhin, 1989), CENDL (INDC, 1991; IAEA, 1996), FOND (Koscheev et al., 2001), ENDFB-VI Release 8 (Lemmel et al., 2001), JENDL-3.2 (Nakagawa et al., 1995; IAEA, 1994), JEF-2.2 and JEFF-3.1 (Nordborg and Salvatores, 1994; WLUP, 2005) for thermal reactor calculations. We have been using the available D4 version of the code for static reactor calculations to investigate differences in reactor parameters arising in a clean core loaded with all fresh elements due to differences in cross section data for various newly released evaluated libraries (Ahmad et al., 2004, 2006; Ahmad and Ahmad, 2005). In fresh fuel there are only a few isotopes (i.e. uranium isotopes in our case) present in fuel regions, and the analysis based on only fresh fuel does not contain the effect of upgraded cross sections of a large number of isotopes contained in various libraries as can be seen in Table 1 (Taubman, 1975; Aldama et al., 2003). The D4 version is unable to evaluate burnup using WLUP libraries due to the addition of many burnup related materials in these libraries (Kulikowska, 2000). There was an intense need to enable the code to use newly evaluated cross sections for depletion analysis. The code needed extensions in fixed dimension arrays to use newly released libraries with increased number of resonant isotopes, burnup nuclides and fission fragments which were limited in the code to use only libraries similar to ‘‘1981’’ WIMS library. For this reason the code was upgraded and named as WIMS-D4S and validated for $^{235}$U depletion analysis by comparing with the results of the newer version of the code WIMS-D5B reported by Aldama et al. (2003). The SARC thus has the capability to use various WIMS-D multigroup libraries i.e. 1981, 1986 and newly released 69 energy groups WLUP libraries containing different number of materials as given in Table 1. Depending on a given library, isotopic composition of the nuclides contained in the library can be computed for a given burnup.

The full core neutronics calculations are strongly dependent on the location of fuel zones and structural materials, i.e. the geometry of the core. In order to do these calculations, the input parameters required in full core modeling by CITATION, such as few group constants of each zone of fuel element, energy released per fission and the fission spectrum, are supplied by processing WIMS-D4S output. Burnup distribution in the core is not uniform due to the different power at which each element is undergoing burnup, and this power may change at each burnup step. Moreover, in the fresh fuel, there are a small number of nuclides in any fuel region, but as fuel undergoes burnup, many fission fragments and actinides are produced which strongly affect the reactivity of the system. For accurate calculations, concentration of these isotopes should be considered in the lattice cell analysis. In any burnup step, the concentrations of different isotopes in fuel at end of the previous step are taken as initial isotopic composition. For this reason, group constants and material composition for each fuel region as a function of burnup are calculated using WIMS-D4S at each burnup step. The composition is used in the following burnup calculations by WIMS-D4S while the group constants are used to solve diffusion equation for integrated group fluxes, excess reactivity and power profile of the whole core using CITATION. Small time steps in burnup can be chosen to achieve accuracy in changes in fuel composition and group constants with burnup. On the other hand, a larger step size is used to achieve results more rapidly in converging parameters. The flow diagram of the SARC is given in Fig. 1. The system is capable of searching equilibrium core for a given reshuffling procedure.

The input of SARC contains the number of fuel elements and their fuel contents at BOC, the composition of different zones, their volumes, burnup step in days, maximum burnup days, maximum allowed discharge burnup of an element, end of cycle reactivity, the reshuffling procedure if more than one cycle is required, number of cycles to burn, convergence criteria for equilibrium core and the specifications for non-fuel regions in the core.

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