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Burnup estimation for plate type fuel assembly of research reactors through the least square fitting method

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ABSTRACT

This work is focused on estimation of burnup for a plate type fuel assembly of research reactors with the SCALE6 code sequences such as TRITON/NEWT and ORIGEN-ARP. And a simple and accurate model is proposed to calculate burnup based on the least square fitting method without additional depletion analyses. One fuel assembly is modeled and its burnup is obtained for different power densities, enrichments, and fuel densities. Linear and non-linear polynomial fitting methods are used to provide a suitable formula for the burnup of the plate type fuel assembly with a function of different parameters. And the proposed approach is applied to three configurations of research reactors, such as a 5 MW core of U_3Si_2 fuel, 3 MW cores with U_3Si_2 and UMo fuels. The results are also validated by comparing those of Monte Carlo codes.

1. Introduction

In order to evaluate the fuel performance and characteristics in the reactor, it is very important to estimate accurately the fuel discharge burnup. Fuel burnup is defined as the amount of energy (usually heat) generated per metric ton of all uranium and plutonium isotopes contained in the fuel charged into a reactor (Kulikowska and Szezesna, 1984). And it is an important quantity for design and operation of reactors from a standpoint of safety as well as operability (Inoue et al., 1969).

It is widely known that the uranium oxide fuel in normal commercial light water reactors approaches the 40–60 GWD/MTU. However, in the research reactor, the burnup is changeable due to different types of fuels such as U_3Si_2, U_3Si, U-Al, and U-Mo fuels. Generally, the metal fuels of research reactors provide higher burnup than the existing uranium oxide fuel of LWR. The discharge burnup is nearly 100 GWD/MTU, which is mainly resulting from the higher power density in research reactors. Therefore, it is important work to estimate accurately the fuel burnup for the safety analysis and the fuel performance analysis. (Ravnik, 1992; Hussein et al., 2011)

In this study, linear and non-linear formulae for burnups of plate type fuel assemblies are suggested with the least square fitting method. This approach enables us to estimate burnup directly without following the detail fuel history including depletion analysis. Accurate burnup estimation is not an easy job due to several reasons such as the effect of fission products and the power change caused by refueling and depletion. In general, power density, uranium enrichment, and fuel density are key factors of the fuel burnup. The sensitivity of each factor has been investigated, and then their effects are combined into one fitted formula for each burnup step.

Several code systems are used to estimate the discharge burnup such as SCALE6 (Bowman, 2007) code system including TRITON/NEWT (Dehart, 2009), ORIGEN-ARP (Gauld et al., 2009a) and Monte Carlo codes such as McCARD (Shim et al., 2012) and MCNPX (Bratton, 2012; Pelowitz, 2010).

The ORIGEN-ARP is the SCALE6 depletion analysis sequence used to perform point-depletion calculations with the ORIGEN-S (Gauld et al., 2009b) code by using problem-dependent cross sections. The NEWT (Dehart, 2005) computer code is a multigroup discrete-ordinates radiation transport code with flexible meshing capabilities that allow two-dimensional (2-D) neutron transport calculations using complex geometric models. The TRITON (Dehart, 2009) is a SCALE control module that enables depletion calculations to be performed by coordinating iterative calls between cross-section processing codes, NEWT, and the ORIGEN-S point-depletion code. The McCARD (Shim et al., 2012) is a Monte Carlo (MC) neutron-photon transport simulation code. It is capable of performing the whole core neutronics calculations, the reactor fuel

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Nomenclature

P	power density (MW/MTU)
E	enrichment (wt%)
D	fuel density (g/cc)
B	percentage burnup
f_{Bu}	percentage burnup predicted value
$n(t)$	U^{235} concentration at time t
S	the sum of square residuals
n	the number of data points
V	variance
m	the number of adjustable parameters (equation coefficients)
r_i	the residual of i-th component

Greek symbol

σ	standard deviation

burnup analysis, the few group diffusion theory constant generation, sensitivity and uncertainty (S/U) analysis, and uncertainty propagation analysis. The MCNPX (Pelowitz, 2010) is a general-purpose Monte Carlo N-Particle code that has been developed as an extension of the MCNP code including the depletion module.

In the Section 2, the procedure of our approach is provided and the general least square fitting equation of our approach is shown in Section 3. Section 4 deals with the application to the research reactor by comparing Monte Carlo code results. Application results and discussions are provided in Section 5. Finally, the conclusion is provided in Section 6.

2. Procedure of calculation

A crude formula is obtained to estimate burnup for plate type fuel assemblies based on different parameters such as power densities, enrichments and fuel densities. To obtain burnup formula, a sensitivity of each factor on burnup is carried out for various time steps where only a specific factor is changed and the others are maintained as constants.

As a first step, the TRITON/NEWT is used to generate burnup dependent cross section library for a plate type fuel assembly. Then the ORIGEN-ARP is used with the obtained to deplete fuel plate for several cycles. From the ORIGEN-ARP analysis the percentage burnup of U-235 is calculated cycle by cycle, and a schematic diagram for the analysis procedure is shown in Fig. 1. The same approach is applied to various cases but using different fuels and time steps.

The percentage burnup is defined as:

$$B = \left(\frac{n(0) - n(t)}{n(0)}\right) * 100\% \tag{1}$$

where $n(0)$ is the initial concentration of U-235 and $n(t)$ is the U-235 concentration at time t.

3. Least square fitting method

The production of fission products and the change in power due to depletion and fuel shuffling after each cycle may disturb a linear relation between the burnup and some parameters. Thus, the first and the second least square fitting methods including the exponential fitting model are tried.

3.1. The first order fitting approach

Our desired equation would be in the linear form such as:

$$f_{Bu} = a_0 + a_1 P + a_2 E + a_3 D \tag{2}$$

where P is a power density (MW/MTU), E is an enrichment (%U-235), and D denotes a fuel density (g/cm^3). To find the coefficient (a), it needs to solve the linear algebra system as follows:

$$\begin{bmatrix} 1 & P_1 & E_1 & D_1 \\ 1 & P_2 & E_2 & D_2 \\ 1 & P_3 & E_3 & D_3 \\ 1 & P_4 & E_4 & D_4 \\ \vdots & \vdots & \vdots & \vdots \end{bmatrix} \begin{bmatrix} a_0 \\ a_1 \\ a_2 \\ a_3 \end{bmatrix} = \begin{bmatrix} B_1 \\ B_2 \\ B_3 \\ B_4 \\ \vdots \end{bmatrix} \tag{3}$$

Rewriting in the symbolic form,

$$Xa = B \tag{4}$$

Finally, the coefficients are obtained through the least square method by multiplying the transpose matrix on both sides of Eq. (4).

$$a = (X^T X)^{-1} X^T B \tag{5}$$

The variance of the j-th parameter, denoted by $V(a_j)$, is estimated as:

$$V(a_j) \approx \frac{S}{n-m}\left(\left[X^T X\right]^{-1}\right)_{jj} \tag{6}$$

The denominator $(n–m)$ is the statistical degrees of freedom and n is the number of data points, and m is the order of fitting number. The residue sum (S) is defined as:

$$S = \sum_{i=1}^{n} r_i^2 \tag{7}$$

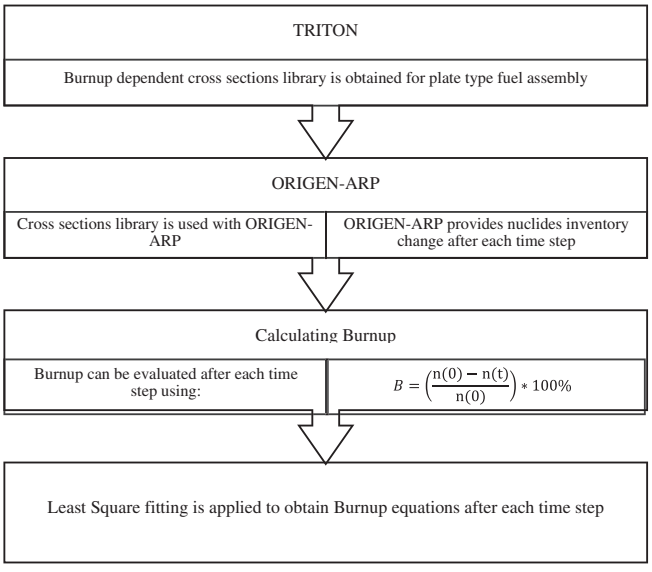

Fig. 1. Procedure of the fitting method for burnup estimation.