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Estimating Burnup for UMo Plate Type Fuel with Least Square Fitting

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1. Introduction

In the previous work,[1] we suggested a new method to estimate burnup for a research reactor with U_3Si_2 fuel through finding fitted equations with SCALE code system.[2]~[4] These equations depended on power density (MW/MTU), uranium enrichment (wt%) and fuel density (g/cc). The objective of this paper is to apply the previous approach to a different core with UMo plate type fuel. The feasibility test of this approach has been done by comparing the results with a Monte Carlo code results.

UMo fuel is a promising candidate for a high performance research reactor and provides better fuel performance including an extended burnup and swelling resistance.[5] Additionally, its relatively high uranium content provides high power density. However, when irradiating UMo fuel in the core, lots of pores are produced due to an extensive interaction between the UMo and Al matrix. The pore leads to an expansion of fuel meat and may result in a fuel failure after all. This problem has almost been solved by using an optimal Si additive to depress the interaction layer. An international program has been performed to manufacture a robust UMo fuel. However, in terms of neutronics, the absorption cross section of Mo is much higher than that of Si, and thus a slightly high uranium density of UMo fuel is required to provide equivalent characteristics to U_3Si_2 fuel. Recently, Korea considers U-Mo fuel for the KJRR design, which is under design stage.

2. Problem Specification

The assumed MTR core consists of 9 fuel assemblies of plate type fuel in 3x3 arrangements and a total power of 3MW. It is beryllium reflected at all sides except at top and bottom and moderated with light water. The core is shown in Fig.1 which is generated using the MCNP visual editor. Each fuel assembly consists of 21 fuel plates. Table 1 shows the composition of U-7Mo fuel.

In this study, several code systems are used to estimate discharge burnup such as TRITON[2], ORIGEN[3] and MCNPX[6].

TRITON is a SCALE control module that carries out depletion calculations to be performed by coordinating iterative calls between cross-section processing codes, NEWT[4]. TRITON/NEWT code is used to generate a database of cross sections for depletion calculation with ORIGEN-ARP. The MCNPX is a general-purpose Monte Carlo N-Particle code that has been developed as an extension of the MCNP code. CINDER90 depletion code was integrated into the latest version of MCNPX to provide it with built-in burnup capabilities.

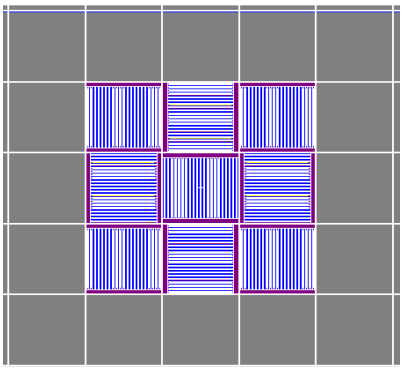

Figure 1. 3x3 MTR core generated by MCNPX

Table1. Isotopic composition of U-7Mo fuels

U-7 Mo Fuel	***Isotope***
	wt%
U-234	1.09E-01
U-235	1.36E+01
U-236	1.52E-01
U-238	5.51E+01
Mo	5.20E+00
Al	2.58E+01
Total	100
U density	5 gU/cc
Fuel density	7.234 g/cc
U:Mo	93:7

3. Methods

First we use TRITON to generate burnup dependent cross section library for the fuel assembly. Then ORIGEN-ARP uses this library to deplete UMo fuel plate. Hence we get the percentage burnup of U-235 cycle by cycle.

Three main parameters are also considered to determine burnup such as enrichment, specific power, and fuel density. In the first case, we change the specific power while the other parameters are constant. The additional two calculations have been done with same assumptions. Many perturbed cases are considered for each calculation. Then we can obtain sensitivity parameters for three variables.

Linear and second order least square fitting were performed. The linear equation would be in the form

$$f_{Bu} = a_0 + a_1 p + a_2 w + a_3 d \qquad (1)$$

where $p = \text{specific } power(\frac{MW}{MTU})$,
$w = enrichment(wt\%)$ and $d = fuel\ density(\frac{g}{cc})$.
To find the coefficients, $a's$, the following matrix form of system is derived.

$$\begin{bmatrix} 1 P_1 E_1 D_1 \\ 1 P_2 E_2 D_2 \\ 1 P_3 E_3 D_3 \\ 1 P_4 E_4 D_4 \\ \vdots\ \vdots\ \vdots\ \vdots \end{bmatrix} \begin{bmatrix} a_0 \\ a_1 \\ a_2 \\ a_3 \end{bmatrix} = \begin{bmatrix} Bu_1 \\ Bu_2 \\ Bu_3 \\ Bu_4 \\ \vdots \end{bmatrix} \quad (2)$$

Or in the symbolic form:

$$Xa = Bu \quad (3)$$

Applying least square fitting method, then

$$a = (X^T X)^{-1} X^T Bu \quad (4)$$

Second order fitting has been done using almost the same procedure, but this time our desired equation is in the form

$$f_{Bu} = a_0 + a_1 P + a_2 P^2 + a_3 E + a_4 E^2 + a_5 D + a_6 D^2 \quad (5)$$

To apply this method we need to know what the power density is for each fuel assembly, this was gotten from Nodal Method calculation that was done for this core[7]; Fig. 3 shows the power density after 100 days of depletion.

FA1	FA2	FA3	
159.06	155.93	159.06	
FA4	FA5	FA6	
155.93	149.61	155.93	
FA7	FA8	FA9	FA ID
159.06	155.93	159.06	MW/MTU

Figure 3. Power density of UMo fuel

4. Results

The first and the second order least square fitting were applied to obtain two fitted equations to estimate burnup after 100 days of depletion and the results were compared to the results of MCNPX.
The obtained least square fitted equations for burnup is

$$Bu = (7.73E+00) + (5.82E-02)P - (3.95E-01)w + (1.47E-02)D \quad (6)$$

$$Bu = (1.57E+01) + (6.00E-02)P - (7.54E-06)P^2 - (1.34E+00)w + (2.47E-02)w^2 + (2.36E-01)D - (1.29E-02)D^2 \quad (7)$$

Where p: specific power (MW/MTU), w: enrichment (wt %), d: density (g/cc). The results compared to MCNPX results are shown in Fig. 4. The maximum error is less than 10%.

5. Conclusions

This work is focused on calculating burnup for plate type UMo fuel through a couple of code systems such as TRITON/NEWT and ORIGEN-ARP. The estimated burnup is compared with that of MCNPX calculation. It is founded that the fitted burnup agrees well with the MCNPX results. This approach will be applicable to easily estimate discharge burnup in research reactor without additional burden. However, some sensitivity tests required for another parameters in order to obtain burnup exactly.

FA1	FA2	FA3	
10.14	9.83	10.06	
-8.49	-7.39	-7.76	
-8.69	-7.58	-7.97	
FA4	FA5	FA6	
9.79	9.23	9.63	
-7.01	-5.41	-5.48	
-7.21	-5.58	-5.68	
FA7	FA8	FA9	FA ID
9.79	9.23	9.63	MCNPX (%U-235)
-5.15	-1.43	-3.59	1st order error(%)
-5.36	-1.63	-3.80	2nd order error (%)

Figure 4. Percent errors of burnup for UMo core

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