Assembly-Based Lattice Characteristics of U-7Mo and U_3Si_2 Plate Fuels for a Research Reactor

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1. Introduction

U-Mo fuel is a promising candidate for a research reactor design and provides better fuel performance including an extended burnup and swelling resistance. Additionally, its relatively high Uranium content provides high power density. However, when irradiating U-Mo fuel in the core, lots of pores are produced due to an extensive interaction between the U-Mo and Al matrix. The pore leads to an expansion of fuel meat and results in a fuel failure after all. During last ten years, many researchers have tried to solve the intrinsic problem of a U-Mo fuel through the international cooperation and it has almost been solved by using an optimal Si additive to depress the interaction layer and an international program has been performed to manufacture a robust U-Mo fuel[1]. 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 U-Mo fuel is required to provide an equivalent level to U_3Si_2 fuel. It is also published that the core performance difference with silicide and molybdenum fuel types is small[2].

To review and make clear the neutronics characteristics of U-Mo fuel, assembly-based lattice calculations have been carried out using the TRITON code.[3] An infinite k value and temperature coefficients are compared as a function of a burnup.

2. Multiplication Factor Comparison

TRITON code carries out two-dimensional neutron transport and depletion calculations. It is also used to provide automated, problem-dependent cross section processing for ORIGEN-ARP in the SCALE-6 code system. It was originally built around NEWT code, which provides the capability to perform a rigorous deterministic transport analysis for a wide variety of problem types[3].

The fuel compositions of U_3Si_2 and U-7Mo fuels are provided in Table I. The enrichment of U-235 is 19.75 wt% and the uranium density of U_3Si_2 fuel is fixed at 4.8 gU/cc, which is a typical value in the design of research reactors. The uranium density of U-7Mo fuel varies from 5 gU/cc to 9 gU/cc.

It has been noted that the weight fraction of Al of U_3Si_2 is similar to that of U-7Mo, but considering the fuel density, more about 20% Al is added the U-7Mo fuel. However the Uranium densities are similar for two fuels, and it is expected that the neutronic characteristics will also be similar for two fuels. The fuel plate is designed with the following data

- Fuel meat thickness : 0.51 mm
- Cladding thickness : 0.38 mm
- Water gap : 2.35 mm
- Cladding material : AG3NE
- Plate number : 21
- Boundary condition : All reflective
- Fuel temperature : 50 ^oC
- Coolant temperature : 40 ^oC (ρ=0.9922 g/cc)
- ENDF/B-VI.8 Library 238-group
- Specific Power: 274.1 MW/MTU
- Burnup : 98.7 GWD/MTU

Figure 1 depicts the k-inf for the U_3Si_2 and U-7Mo fuels including various uranium densities of U-7Mo fuel. As expected, the depletion behavior is similar for all cases. Figure 2 shows the reactivity difference between the U_3Si_2 and U-7Mo fuels. When increasing the uranium density of U-Mo fuel, the reactivity difference increases up to 13 mk and decreases slightly as the burnup proceeds.

Table I. Isotopic Composition of U_3Si_2 and U-7Mo Fuels

Isotopes	wt%	Isotopes	wt%
U-234	1.072E-01	U-234	1.090E-01
U-235	1.340E+01	U-235	1.362E+01
U-236	1.494E-01	U-236	1.519E-01
U-238	5.419E+01	U-238	5.510E+01
Si	9.076E+00	Mo	5.203E+00
Al	2.308E+01	Al	2.583E+01
Total	100	Total	100
U density	4.8 gU/cc	U density	5.0 gU/cc
Fuel Density	6.732 g/cc	Fuel Density	7.234 g/cc
U:Si	88.2:11.8	U:Mo	93:7

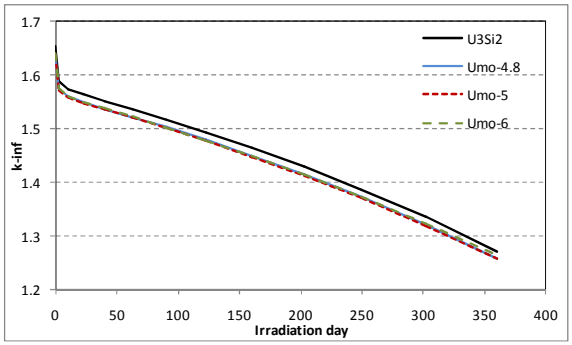

Figure 1. K-inf for as a function of irradiation

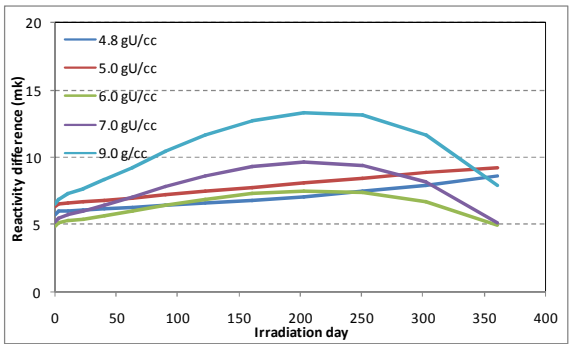

Figure 2. Reactivity difference between U_3Si_2 and U-7 Mo fuels

3. Temperature Coefficients

The fuel temperature coefficient (FTC) and coolant temperature coefficient (CTC) are obtained from the following perturbation conditions. In the TRITON code, the perturbation option is useful with the depletion calculations.

- Perturbed fuel temperatures : 45 ^oC, 55 ^oC

-Perturbed coolant temperatures : 35 ^oC (ρ=0.9940 g/cc), 45 ^oC(ρ=0.9902 g/cc)

-Uranium density of U-7Mo : 5 gU/cc

Figures 3 and 4 show the fuel and coolant temperature coefficients as a function of irradiation time, respectively. In the case of FTC, U-7Mo fuel provides 3 times lower than that of U_3Si_2 fuel, which results from a 7% higher uranium content in U-7Mo fuel, as shown in Table 1. Furthermore, the CTC of U-7Mo fuel is also about 10% less negative than that of the U_3Si_2 fuel. After 360 irradiation days, the FTCs of U_3Si_2 and U-7Mo fuels are -0.035 mk/^{o}C and -0.112 mk/^{o}C, respectively. The CTCs after 360 days are -0.035 mk/^{o}C and -0.039 mk/^{o}C for the U_3Si_2 and U-7Mo fuels, respectively. Therefore, there is a greater safety margin when U-7Mo fuel is taken into consideration.

4. Conclusions

Lattice depletion analyses for U-7Mo fuel have been carried out with various uranium densities to find an equal U_3Si_2 fuel. It has found that about 5 gU/cc of U-7Mo fuel is equal to 4.8 gU/cc of U_3Si_2 fuel considering the depletion behavior. The temperature feeedback coefficients are also calculated as a function of irradiation time. It was found that the U-7Mo fuel provides better characteristics not only in neutronics but also in a safety analysis. In the near future, U-Mo fuel will be investigated for the realistic application of a core analysis including a full core analysis.

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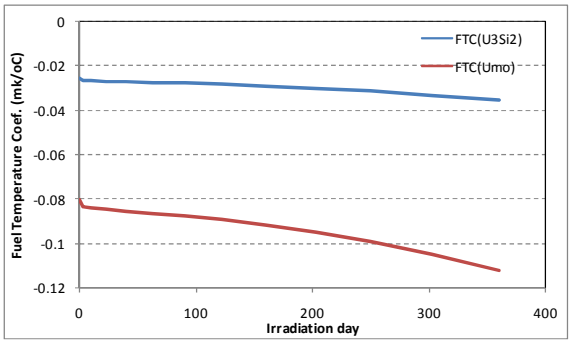

Figure 3. Fuel temperature coefficient for U3Si2 and U-7Mo fuels.

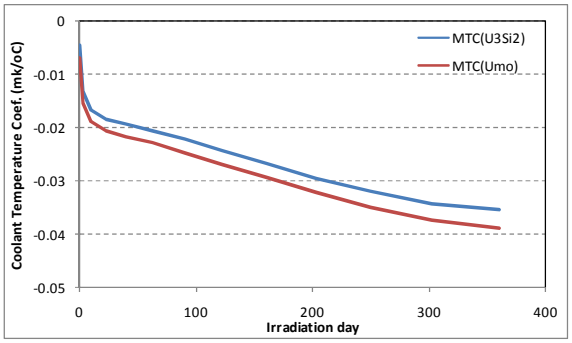

Figure 4. Coolant temperature coefficient for U3Si2 and U-7Mo fuels