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LATTICE CHARACTERISTICS AND ACTIVITY ANALYSIS OF U_3SI_2 AND UMo PLATE TYPE FUEL ASSEMBLIES WITH SCALE6 CODE

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ABSTRACT

SCALE6 code is used to analyze lattice characteristics and source term analysis for U_3SI_2 and UMo fuels. Among lattice characteristics, reactivity change and temperature feedback coefficients such as moderator temperature coefficient (MTC) and fuel temperature coefficient (FTC) are calculated for two kinds of fuel types with the TRITON/NEWT module in SCALE6 code. By using the reactor dependent libraries from TRITON code, the burnup dependent activity analysis is also carried out in the ORIGEN-ARP module for plate type fuels. The activity of several long lived fission products such as Tc-99, Zr-93, I-137 are obtained from the results, which needs careful treatment in the spent nuclear fuel. Furthermore, neutron and gamma spectra are evaluated at various cooling periods, which are main input data for the radiation shielding analysis for research reactors. Additionally, some sensitivity analyses are performed for several different uranium contents in UMo fuel.

1. Introduction

A new research reactor of 15 MW, Ki-Jang research reactor (KJRR), is under being designed in Korea, which uses UMo plate type fuel assemblies. The reactor will be dedicated to medical and industrial purposes such as radioisotope production and neutron transmutation doping. UMo fuel is a promising candidate for a high performance research reactor 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 UMo fuel in the core, lots of pores are produced due to an extensive interaction between the UMo and Al matrix[1]. The pore leads to an expansion of fuel meat and may result in a fuel failure after all. During last ten years, many researchers have tried to solve the intrinsic problem of a UMo fuel through the international cooperation and it 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[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 UMo fuel is required to provide an equivalent characteristics to U_3Si_2 fuel. It is also published that the core performance difference with silicide and moly fuel types is small[2]. To review and make the neutronics characteristics of UMo fuel clear, assembly-based lattice calculations have been carried out using the TRITON/NEWT code.[3][4]
Some results for temperature feedback coefficients such as moderator temperature coefficient (MTC) and fuel temperature coefficient (FTC) are shown in Section 2. And k-infinite values are also compared as a function of burnup.

Section 3 deals with the burnup dependent activity analysis using the ORIGEN-ARP module. The reactor dependent libraries are obtained from the results of TRITON/NEWT code. Additionally, the activity changes of several long lived fission products such as Tc-99, Zr-93, I-137 are provided because they needs special treatment when processing the spent nuclear fuel. Neutron and gamma spectra are also evaluated at various cooling periods, which become main source term for the radiation shielding analysis. Finally, the summary of this study is provided in Section 4.

2. Temperature Feedback Coefficients

In order to obtain temperature coefficient, the TRITON/NEWT 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[5] in the SCALE-6 code system. The NEWT code provides a rigorous deterministic transport analysis for a wide variety of problem types[4].
The fuel compositions of U_3Si_2 and U-7Mo fuels are provided in Table 1. The enrichment of U-235 is 19.75 wt%. 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, whereasthe uranium density of U-7Mo fuel varies from 5 gU/cc to 8 gU/cc for sensitivity test.
It has been noted that the weight fraction of Al in U_3Si_2 fuel is similar to that in U-7Mo fuel, but considering the fuel density, about 20 % Al more is added the U-7Mo fuel. In case of similar uranium density for two types of fuels, 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
- Number of plate : 21
- Boundary condition : All reflective
- Fuel temperature : 50 ^{o}C
- Coolant temperature : 40 ^{o}C (density=0.9922 g/cc)
- ENDF/B-VI.8 Library 238-group
- Specific Power: 274.1 MW/MTU
- Burnup : 98.7 GWD/MTU

Table 1. Isotopic composition of U_3Si_2 and U-7Mo fuels

Isotope	U_3Si_2 Fuel	Isotope	U-7Mo Fuel
	wt%		wt%
U-234	1.163E-01	U-234	1.090E-01
U-235	1.454E+01	U-235	1.362E+01
U-236	1.621E-01	U-236	1.519E-01
U-238	5.881E+01	U-238	5.510E+01
Si	5.970E+00	Mo	5.203E+00
Al	2.040E+01	Al	2.583E+01
Total	100	Total	100
U density	4.8 gU/cc	U density	5.0 gU/cc
Fuel Density	6.519 g/cc	Fuel Density	7.234 g/cc
U:Si	92.5:7.5	U:Mo	93:7

Fig. 1 depicts the k-infinite for the U_3Si_2 and U-7Mo fuels with various uranium densities from 5 gU/cc to 8 gU/cc. As expected, the depletion behavior is similar for all cases. The reactivity difference between U_3Si_2 and U-7Mo fuels is less than 10 mk. The U_3Si_2 fuel provides a slightly higher reactivity due to less uranium loading even if the fuels are irradiated at the same power density.

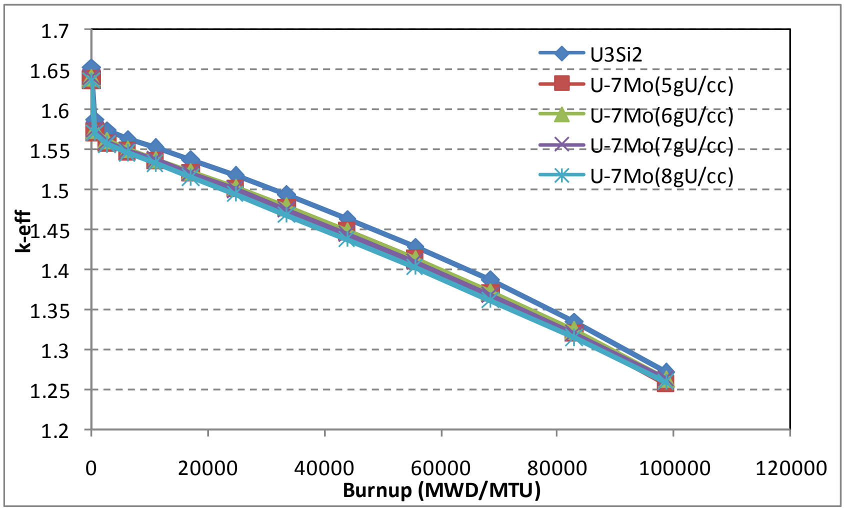

Fig. 1. k-infinite change as a function of burnup

The fuel temperature coefficient (FTC) and moderator temperature coefficient (MTC) are obtained from the following perturbation conditions. In the TRITON/NEWT code, the perturbation option is useful with the depletion calculations.
-Perturbed fuel temperatures : 45 ^{o}C, 55 ^{o}C
-Perturbed moderator temperatures : 35 ^{o}C (density=0.9940 g/cc), 45 ^{o}C(density=0.9902 g/cc)
-Uranium density : U_3Si_2 fuel = 4.8 gU/cc, U-7Mo fuel = 5 gU/cc

Figs. 2 and 3 show the fuel and moderator temperature coefficients as a function of irradiation time, respectively. In the case of FTC, U-7Mo fuel provides slightly lower than that of U_3Si_2 fuel, which results from a 4% higher uranium density and 30% higher Al contents in U-7Mo fuel, as shown in Table 1. Furthermore, the MTC 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 -0.037 mk/^{o}C, respectively. The MTCs 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 considered as a fuel for research reactor. Table 2 shows the FTC as a function of the irradiation day.

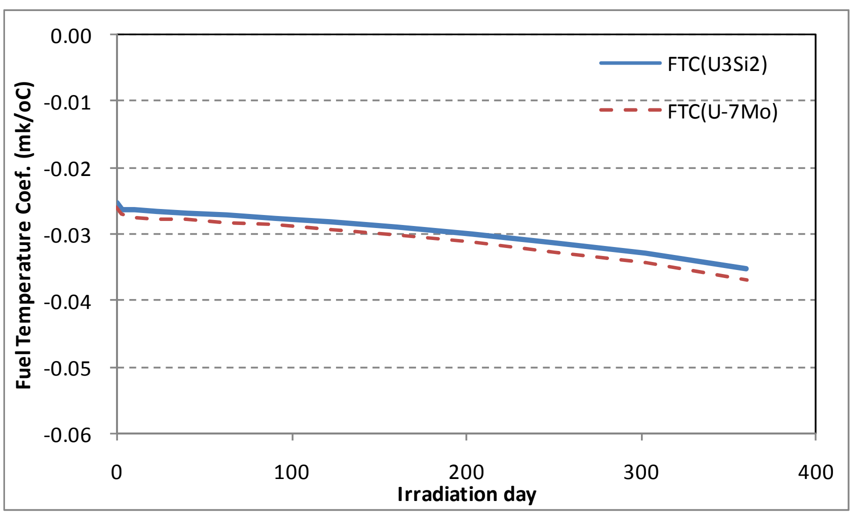

Fig. 2. Fuel temperature coefficient for U_3Si_2 and U-7Mo fuels.

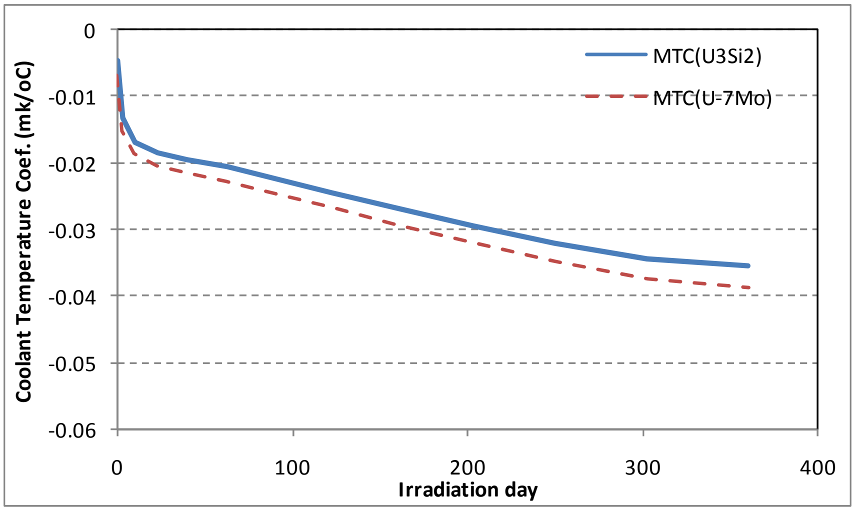

Fig. 3. Moderator temperature coefficient for U_3Si_2 and U-7Mo fuels.

Table 2. FTC for irradiation days

Day	MTC (mk/ °C)		
	U_3Si_2	U-7Mo	Diff. (U_3Si_2 - U-7Mo)
0	-0.02530	-0.08025	0.05495
2.5	-0.02531	-0.08025	0.05702
10	-0.02623	-0.08325	0.05764
22.5	-0.02645	-0.08409	0.05803
40	-0.02663	-0.08466	0.05849
62.5	-0.02685	-0.08534	0.05929
90	-0.02718	-0.08646	0.05984
122.5	-0.02760	-0.08745	0.06105
160	-0.02817	-0.08922	0.06282
202.5	-0.02893	-0.09175	0.06516
250	-0.02994	-0.09510	0.06814
302.5	-0.03123	-0.09937	0.07188
360	-0.03297	-0.10485	0.07701

3. Fuel Irradiation Analysis

It is required to estimate fuel inventory, activity, decay heat, and neutron and gamma spectra when fuel is loaded in a reactor. In general, the ORGIEN-S code is widely used due to its capability of huge number of isotopes and decay chains. To obtain accurate irradiation results, a reactor dependent library is essential. Thus, it is newly made for the research reactor with the plate type fuel assembly by using the TRITON/NEWT code. The irradiation and decay analysis have been carried out with ORIGEN-S code for plate type fuel assembly. Irradiation day is given 360 days and the specific power is assumed as 250 MW/MTU. The cooling time is provided 1 year. Tables 3 and 4 show the activity and decay heat of the spent fuel at different cooling times, respectively. It is found that U-7Mo fuel provides slightly higher activity and decay heat than U_3Si_2 fuel, which results from the uranium content and the activation of Mo. After 1 year cooling, the U-7Mo fuel provides about 3% higher activity and decay heat.

Table 3. Activity of U_3Si_2 and U-7Mo(5 gU/cc) fuels (unit: Ci)

Isotope	U_3Si_2		U-7Mo (5 gU/cc)	
	Discharge	1 year	Discharge	1 year
Cs-137	2.99E+05	2.92E+05	2.98E+05	2.92E+05
Ba-137m	2.82E+05	2.76E+05	2.82E+05	2.75E+05
Y-90	2.95E+05	2.76E+05	2.90E+05	2.72E+05
Sr-90	2.83E+05	2.76E+05	2.78E+05	2.71E+05
Ru-103	7.34E+06	1.17E+04	7.56E+06	1.20E+04
Rh-103m	7.33E+06	1.17E+04	7.54E+06	1.20E+04
Pu-241	8.44E+04	8.04E+04	1.24E+05	1.18E+05
Ce-141	1.20E+07	5.02E+03	1.19E+07	4.97E+03
Pu-238	1.40E+03	1.45E+03	2.51E+03	2.60E+03
Kr-85	3.72E+04	3.49E+04	3.67E+04	3.44E+04
Total	3.39E+08	9.39E+06	3.47E+08	9.46E+06

Table 4. Decay heat of U_3Si_2 and U-7Mo (5 gU/cc) fuels (unit: W)

Isotope	U_3Si_2		U-7Mo (5 gU/cc)	
	Discharge	1 year	Discharge	1 year
Y-90	1.63E+03	1.53E+03	1.61E+03	1.50E+03
Ba-137m	1.11E+03	1.08E+03	1.11E+03	1.08E+03
Cs-137	3.32E+02	3.25E+02	3.32E+02	3.24E+02
Sr-90	3.28E+02	3.20E+02	3.23E+02	3.15E+02
Ru-103	2.44E+04	3.89E+01	8.33E+01	8.62E+01
Pu-238	4.63E+01	4.80E+01	2.52E+04	4.00E+01
Pu-239	1.26E+01	1.29E+01	1.44E+01	1.47E+01
Pu-240	1.02E+01	1.02E+01	1.11E+01	1.11E+01
Ce-141	1.76E+04	7.35E+00	1.74E+04	7.27E+00
Pm-148m	1.44E+03	3.14E+00	1.82E+03	3.97E+00
Total	1.59E+06	3.58E+04	1.61E+06	3.67E+04

Figs. 4 and 5 show the neutron and gamma spectra for U_3Si_2 and U-7Mo fuels at 1 year cooling, respectively. And Table 5 shows the neutron and gamma intensity for various fuel cases. As expected, the neutron and gamma intensity proportionally increases as uranium content in U-7Mo fuel increases. And the ORIGEN libraries are constructed for two different base libraries of ENDF/B-

VI and ENDF/B-VII using SCALE6. From the irradiation test with U_3Si_2 fuel, the difference of libraries is insignificant as shown in Fig. 6.

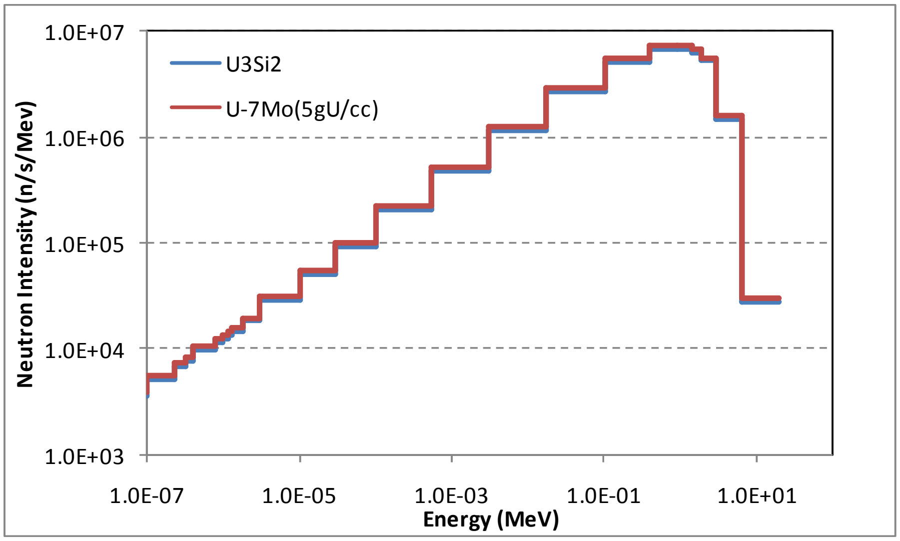

Fig. 4. Neutron spectra for U_3Si_2 and U-7Mo fuels.

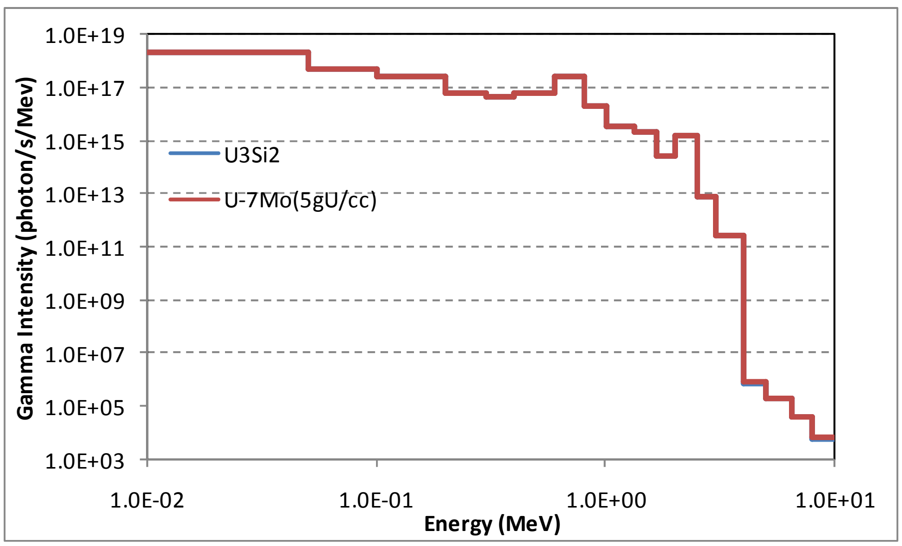

Fig. 5. Gamma spectra for U_3Si_2 and U-7Mo fuels.

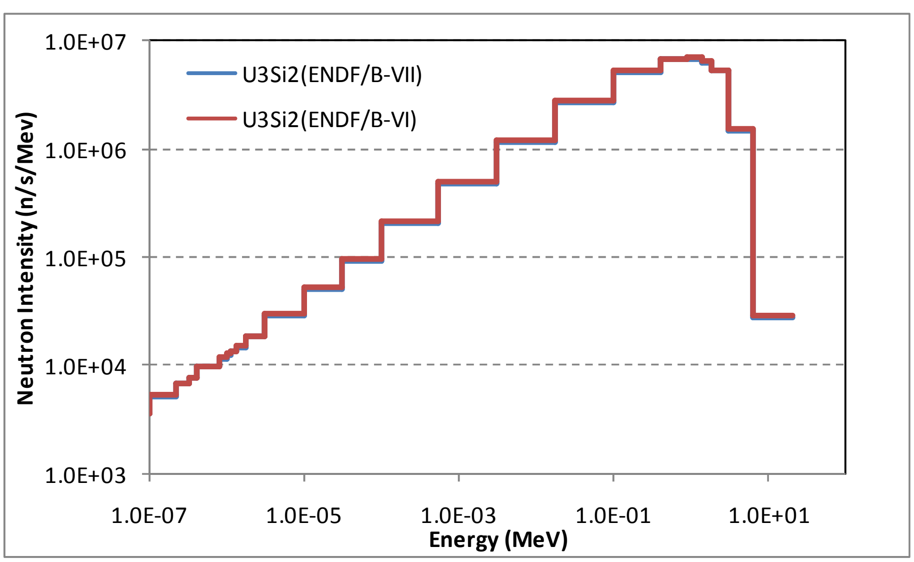

Fig. 6. Neutron spectra for different libraries.

It should be treated carefully for long lived fission products such as Tc-99 (half life = 2.1E+5 year), Se-79 (half life = 1.5E+6 year), and I-129 (half life = 1.6E+7 year), and high decay heat releasing fission products such as Sr-90 and Cs-137. Among fission products, activity changes of several long half-lived fission products are investigated as a function of the irradiation and decay times. Fig. 7 depicts the activity variation for different irradiation times. And Table 6 shows the tabulated data of activity change as a function of decay time up to 100 years. Due to long half life, the activity change is almost invariant as though decay time increases. The difference between U_3Si_2 (4.8 gU/cc) and U-7Mo(5 gU/cc) fuels is not so significant, as expected.

Table 5. Neutron and gamma intensities for various cases

Case	Uranium Density (gU/cc)	Neutron Intensity (neutron/s)	Gamma Intensity (photon/s)
U_3Si_2(ENDF/B-VII)	4.8	2.29E+07	2.14E+17
U_3Si_2(ENDF/B-VI)	4.8	2.34E+07	2.14E+17
U-7Mo	5.0	2.47E+07	2.15E+17
U-7Mo	6.0	3.41E+07	2.17E+17
U-7Mo	7.0	4.34E+07	2.19E+17
U-7Mo	8.0	5.48E+07	2.21E+17
U-7Mo	9.0	6.90E+07	2.23E+17

*ORIGEN-ARP, 360 irradiation, 1 year cooling

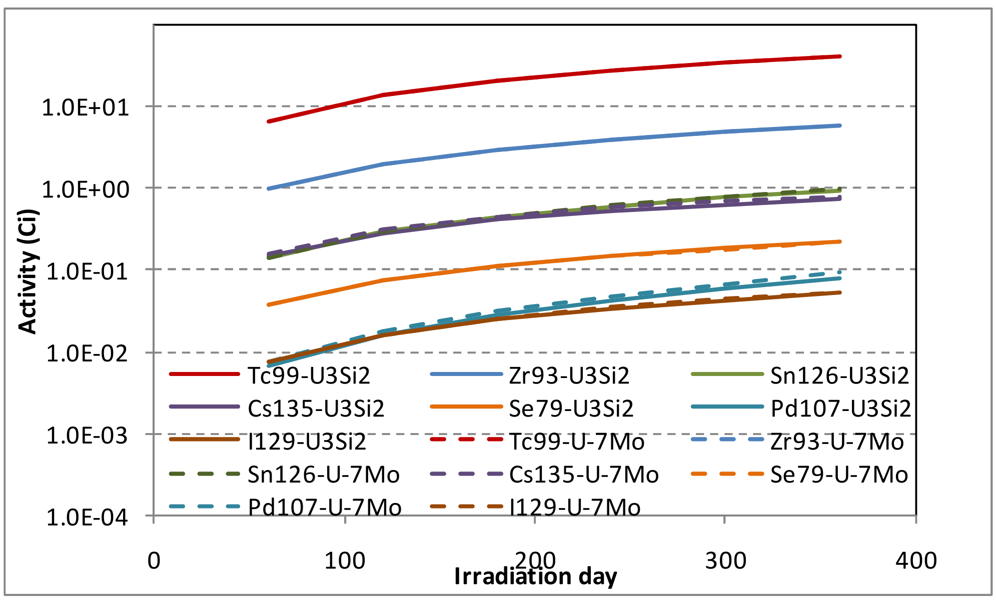

Fig. 7. Activity change for various long-lived fission products.

4. Conclusions

The lattice calculation has been carried out with TRITON/NEWT code for the U_3Si_2 and U-7Mo fuels. The behavior of two different fuels is similar but slightly different in the temperature feedback coefficients. The U-7Mo fuel provides more negative temperature coefficients and different behavior when changing uranium contents. The irradiation analysis is also performed with ORIGEN-ARP code with the libraries obtained from TRITON/NEWT code. The neutron and gamma spectra including time dependent activity and decay heat are also obtained. The general behavior of two different fuels is almost same because similar uranium densities are taken into consideration.

When applying UMo fuel in a research reactor, the uranium density increases up to about 8 gU/cc. UMo fuel with higher uranium density provides longer fuel cycle and different behavior of irradiation characteristics. Therefore, more detail analysis should be followed with the full core analysis in order to compare the neutronics characteristics for UMo fuel.

Table 6. Activity change as a function of decay time (unit: Ci)

Isotope	Half life (10^6 year)	U_3Si_2 (4.8 gU/cc)		U-7Mo (5 gU/cc)	
		Discharge	100 year	Discharge	100 year
Tc-99	2.11E-01	4.06E+01	4.10E+01	3.97E+01	4.02E+01
Zr-93	2.30E-01	5.87E+00	5.87E+00	5.79E+00	5.80E+00
Sn-126	3.27E-01	9.42E-01	9.41E-01	9.68E-01	9.67E-01
Se-79	1.53E+00	2.18E-01	2.18E-01	2.16E-01	2.16E-01
Cs-135	2.30E+00	7.31E-01	7.35E-01	8.02E-01	8.06E-01
Pd-107	6.50E+00	7.97E-02	7.97E-02	9.22E-02	9.22E-02
I-129	1.57E+01	5.29E-02	5.43E-02	5.40E-02	5.54E-02

References

[1] J.M. Park, H.J. Ryu, J.H. Yang, Y.S. Lee, B.O. Yoo, Y.H. Jung, H.M. Kim, C.K. Kim, Y.S. Kim, and G.L. Hofman, “KOMO-4 Test PIE Results”, RERTR 2010 – 32nd International Meeting on Reduced Enrichment for Research and Test Reactors, October 10-14, 2010, Lisbon, Protugal (2010).

[2] R.B. Pond, N.A. Hana, J.E. Matos, and G. Ball, “A Neutronic Feasibility Study for LEU Conversion of the SAFARI-1 Reactor”, RERTR 2000–International Meeting on Reduced Enrichment for Research and Test Reactors, October 1-6, 2000, Las Vegas, U.S.A. (2000).

[3] M.D. DeHart, TRITON: A Two-Dimensional Transport and Depletion Module for Characterization of Spent Nuclear Fuel, ORNL/TM-2005/39, Ver.6, Oak Ridge National Laboratory (2009).

[4] M.D. DeHart, NEWT: A New Transport Algorithm for Two-Dimensional Discrete Ordinates Analysis in Non-Orthogonal Geometries, ORNL/TM-2005/39, Ver.6, Vol. II, Sec. F21, Oak Ridge National Laboratory, 2009.

[5] I.C. Gauld, S.M. Bowman, and J.E. Horwedel, ORIGEN-ARP: Automatic Rapid Processing for Spent Fuel Depletion, Decay, and Source Term Analysis, ORNL/TM-2005/39, Ver.6, Vol. I, Sec. D1, Oak Ridge National Laboratory, 2009.