Volume 16, Number 1, 2023. pp. 91-98

Jordan Journal of Physics

ARTICLE

Neutron-Physical and Radiation Characteristics of Different Low Enrichment Fuels (Thorium and Uranium) in a Proposed JRTR Research Reactor

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Doi: **https://doi.org/10.47011/16.1.9**

Received on: 05/06/2021; *Accepted on: 15/08/2021*

Abstract: The Jordan Research and Training Reactor (JRTR) is a 5-MWth, open-pool type, light-water moderated and cooled reactor with a heavy-water reflector system. The core consists of standard Material Testing Reactor (MTR), the plate type of fuel assemblies with low enriched fuel of 19.75% U-235 enrichment. One of the most important purposes of JRTR is producing several radioisotopes e.g. (^{99}Mo/^{99m}Tc, ^{131}I and ^{192}Ir isotopes) by using the neutron activation technology. A new computational model has been developed for different low enrichment fuels (Thorium and Uranium) in the JRTR by using the Serpent Monte Carlo code. The purpose of this paper is to validate Th-U233 fuel by comparison of the calculation results of important neutronics parameters, like k_{eff}, flux distribution, kinetics parameters, power peaking factors, heat decay and activity of spent fuel.

Keywords: U_3Si_2, Th_3Si_2, JRTR, Light-water Reactors (LWRs), Extended cycle lengths.

Introduction

Uranium silicide, U_3Si_2, is a candidate accident tolerant fuel type to replace uranium dioxide (UO_2) used in light-water reactors (LWRs) [1]. U_3Si_2 is desired for its increased thermal conductivity and higher uranium loading as compared to UO_2 [2], leading to an economic benefit in terms of lower enrichments, extended cycle lengths or power uprates. The economic advantages of U_3Si_2 may also enable the cost-effective adoption of accident tolerant cladding concepts [1]. Industry partners are interested in deploying U_3Si_2 in the current LWR fleet relatively soon and desire a more pure U_3Si_2 fuel pellet than what was previously fabricated [2]. However, due to the corrosion characteristics of U_3Si_2 [3], pure U_3Si_2 fuel pellets will likely not be used unless as a part of a high uranium-density composite.

After the qualification of U_3Si_2-Al dispersion fuel up to the uranium density of 4.8 gU/cm^3 was approved by the US Nuclear Regulatory Commission (NRC) [18] in 1988, more than 40 research reactors have adopted low enriched uranium (LEU) fuel of uranium silicide. Considerable experience in irradiation performance and manufacturing has been accumulated on the U_3Si_2–Al dispersion fuel in research and test reactors. In this regard, U_3Si_2-Al dispersion fuel is presently considered one of the best qualified fuels in terms of uranium loading and performance in these reactors.

One of the most critical problems facing the development of nuclear energy is nuclear waste, which includes the hazards of long-lived isotopes and proliferation risks [4, 5]. Many countries started to study the thorium fuel cycle to share with currently used cycles [6] or as an alternative to them [7].

To save uranium resources, a pure Th-232/233U cycle will indeed produce less

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plutonium and minor actinides. The long-term radiotoxicity of thorium-based spent nuclear fuels is more accurately described as being comparable to that of uranium-based spent nuclear fuels and resisting the dangers of nuclear proliferation. A worldwide interest for the thorium fuel cycle has increased since 1960 [8, 19].

Both U-238 and Th-232 are considered fertile materials, because they absorb neutrons and turn into fissile materials (Pu239 and U233, respectively). Besides, U-233, U-235 and Pu-239 are considered fissile materials, because they possess fission with absorbing a neutron. Th-232 is considered better fertile than U-238 because of the higher absorption cross-section of Th-232 compared to U-238. U-233 is considered better fissile, because it has a lower capture cross-section than U-235 and other fissile nuclides.

The conversion of Th-232 into U233 can occur in both thermal and fast reactors. Among these reactors are high-temperature gas-cooled reactors (HTGRs), light water breeder reactors (LWBRs), pressurized heavy-water reactors (PHWRs), liquid metal cooled fast breeder reactors (LMFBRs) [9], supercritical water reactors (SCWRs) [10, 11] and molten-salt breeder reactors (MSBR) [4, 12, 13]. In India, at Bhabha Atomic Research Center (BARC), high-purity U-233, which contains less than five ppm U-232, was successfully obtained starting from the year 1970 [8].

In the previous mentioned information above, studies about using the two types of nuclear fuel (uranium silicide and thorium fuel) showed an improvement in the performance of nuclear fuels and reached a practical application for each of them in some stations [1, 2, 6 and 7]. In this paper, we will study the combination of these two types of nuclear fuel and study their performance and radiological properties, where we will compare the performances of uranium silicide-U_3Si_2 and thorium silicide-Th_3Si_2 fuels.

The Aim of Work

This work considers one of the fuel assemblies of JRTR reactor to make a comparison between the performances of LEU (uranium silicide-U_3Si_2) and (thorium silicide-Th_3Si_2) fuels without any modifications of the assembly and with the same operation conditions, but with different fissile enrichments.

Model Description

The main characteristics of the fuel assembly are presented in Table 1 and Fig. 1 [14]. All calculations were performed using the Serpent2 code [15], based on the solution of the neutron transport equation by the Monte Carlo method. Nuclear data was obtained from the library of nuclear constants ENDFB7 [16].

TABLE 1. Major parameters of the fuel assembly and fuel plate.

Parameter	Values
Fuel meat	U_3Si_2-Al
U-235 enrichment	19.75 wt%
U density in fuel meat	4.8 g U/cm^3
Fuel-meat density	6.543 g/cm^3
Fuel-meat thickness	0.51 mm
Fuel-plate thickness	1.27 mm
Fuel-plate width	70.7 mm
Fuel-plate length	680 mm
Coolant-channel thickness	2.35 mm
Fuel-assembly width	76.2 mm
Number of fuel plates/fuel assembly	21
Fuel-assembly height	1015 mm
Cladding material	Aluminum alloy
Clad thickness	0.38 mm
Cladding density	2.7 g/cm^3
Average thermal power/FA	0.27778 MW/FA
Maximum temperature of fuel	403 K
Maximum temperature of other materials	313 K

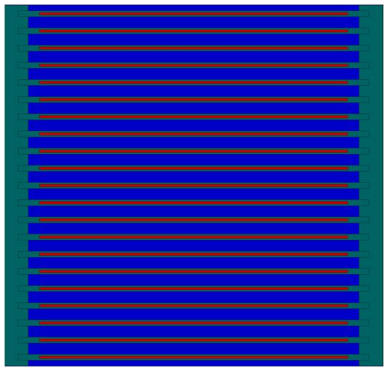

FIG. 1. A top view of the fuel assembly and fuel plate.

Results and Discussion

For all subsequent calculations, the density of the silicide of the fuel (uranium silicide and thorium silicide) was constant and equal to 6.543 g/cm^3. The comparison of fuel performance was under the same operating conditions, where the maximum fuel temperature is 403 K, the cooling light-water temperature is 313 K and its density is 1 g/cm^3.

The first step in our work was to determine the enrichment of U-233 in thorium silicide to obtain the same value of the neutron multiplication factor (K_{inf}) in uranium silicide in 0 burnup and at 100 MW.day/Kg U. For that, in this paper, several fuel variants (V_1-V_4), presented in Table 2, have been studied. In this paper, we called variant V_1 the standard variant.

TABLE 2. The enrichments of the fuels in the considered variants of calculations.

Variant	V_1	V_2	V_3	V_4
Components	U_3Si_2	Th_3Si_2	Th_3Si_2	Th_3Si_2
Enrichment, %	19.75	19.75	9.75	15.75

Neutron's Flux

The spectrum of neutron flux and its distribution, depending on height for all variants, are presented in Fig. 2 (a and b).

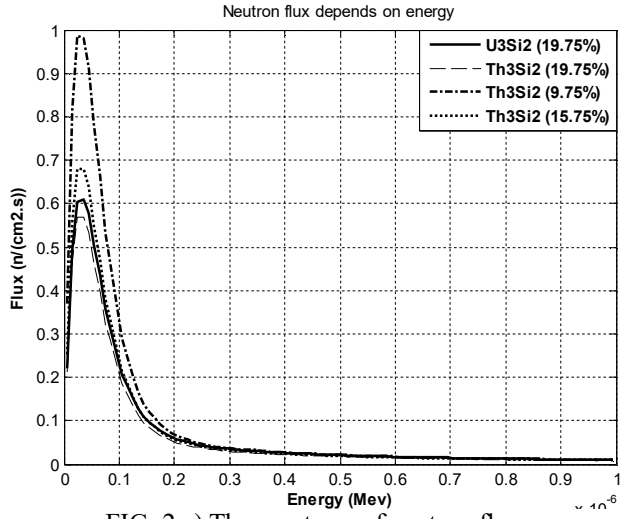

FIG. 2 a) The spectrum of neutron flux.

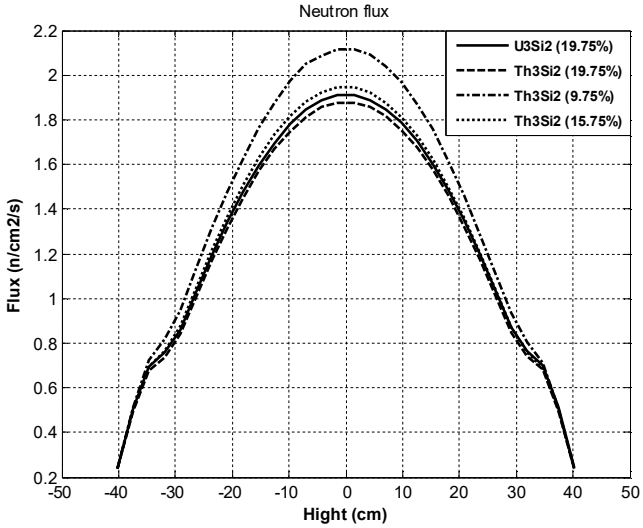

FIG. 2 b) The distribution of neutron flux depending on the height of FA.

From Fig. 2, it is clear that the neutron flux of thorium silicide is less than that of uranium silicide (in the same enrichments (V_1 and V_2)) and this is due to the difference in the absorption cross-section for thermal neutrons of the isotopes of the fuel components. Th-232 has a higher thermal capture cross-section than U-238 (thermal capture cross-section for Th-232 and U-238 is 7.6 and 2.7 barns, respectively). So, more Th-232 will convert into fissile U-233 than U-238 will convert into U-239. [17].

On the other hand, the decrease in thorium enrichment increases the neutron flux and this is a natural result of the decrease in the concentration of fissile isotopes, which have a high ability to absorb thermal neutrons.

Power Distribution

The spatial distribution of energy release (in fuel plates) and the overall radial coefficient of unevenness in fuel assemblies are presented in Table 3.

TABLE 3. The coefficient of uneven distribution of energy release for all variants.

Variant	V_1	V_2	V_3	V_4
The coefficient of uneven distribution of energy release	1.11	1.10	1.11	1.11

As we can see from Table 3, the difference between the coefficients of the uneven energy release distribution for all variants is almost equal with a very slight preference for thorium.

Temperature Coefficients

Under normal operating condition, the range of temperature variation in the JRTR is relatively narrow compared with that in a power reactor; for instance, the coolant temperature rise through the core at full power is only 7°C and the difference in fuel-meat average temperatures at zero power and full power is only about 15°C. For that, the temperature coefficients of reactivity for coolant and fuel under normal operating condition are very small.

Concerning fuel performance, the operating experiences can be observed in the aspect of fuel temperature, cladding corrosion, swelling and dimensional stability. The fuel temperature of the JRTR is estimated to be less than 130°C, whereas full-sized fuel assemblies were irradiated and the performance of the fuel was demonstrated successfully with the maximum fuel temperature up to 130°C in NUREG-1313[18].

Table 4 presents the fuel temperature coefficients of reactivity (403-503) K for the two types of fuel for an uncertainty of criticality of 0.00018, which was achieved by 6000 neutrons with 1020 cycles and with neglecting the first 20 cycles.

TABLE 4. The fuel temperature coefficients of reactivity (403-503) K for the two types of fuel.

Range of temperature	Fuel temperature coefficient of reactivity for U_3Si_2	Fuel temperature coefficient of reactivity for Th_3Si_2
(403-503) K	-1.4402E-04	-9.4617E-05
(403-603) K	-1.8789E-05	-1.0814E-05

It is quite clear that the fuel temperature coefficient of reactivity is negative in both types of fuels and both of its values are also close, which means that the difference in the type of fuel does not affect the kinetics of the reactor and the reason for this is lowering the temperature of the fuel during the cycle.

Burnup

The neutron infinity multiplication factor (K_{inf}) was calculated for JRTR FA in this paper depending on the burnup by using Serpent for all the variants of fuel, as presented in Table 2 and Fig. 3. For all variants, the power was 2.77778E+5 W/FA and the density of fuel was 6.543 g/cm^3.

Our simulation shows that, at the same density of fuel, the enrichment of U-233 in thorium silicide 9.75% (V_3) is sufficient to reach a neutron multiplication factor (K_{inf}) with the same value of uranium silicide with enrichment 19.75% of U-235. Low enrichment of uranium in thorium fuel is due to the exchanging of a fissile U-235 by U-233, which has a lower capture cross-section compared with that for U-235 (capture cross-section for U-233 and U235 are 54 and 100 barns, respectively) and more neutrons are produced per neutron absorbed in fuel (η) or reproduction factor (η for U-233 and U-235 is 2.26 and 2.08 at thermal neutrons, respectively). But, the low fuel enrichment leads to faster decreasing of the neutron multiplication factor depending on the burnup and that is due to lower concentration of fissile isotopes (where the power was constant for all variants).

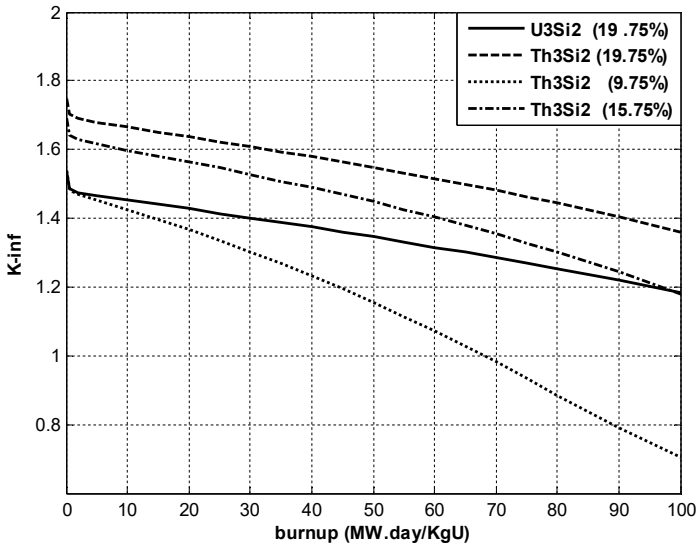

FIG. 3. The neutron multiplication factor K_{inf} *versus* the burnup.

In variant V_4, despite that the enrichment of U-233 in thorium silicide is lower than the enrichment of U-235 in uranium silicide, the neutron multiplication factor continues to be higher than its value in uranium silicide until the end of burnup of fuel.

In variant V_2, where the enrichment of U-233 is 19.75% (equal to the enrichment of U-235 in uranium silicide), the neutron multiplication factor is always higher than the standard variant (V_1) and this is due to the difference in the absorption cross-section of thermal neutrons between the isotopes, as we mentioned earlier.

Cumulative Isotopes

Figs. 4-7 present the mass of isotopes (Pu-239, Cm-244, Th-233 and Cs-134) depending on the time of operation.

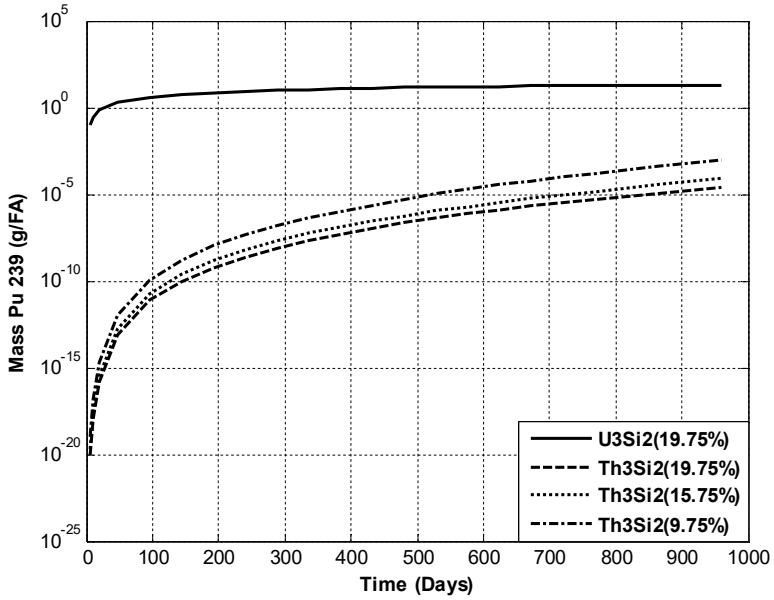

FIG. 4. The mass of Pu-239 depending on the time of operation.

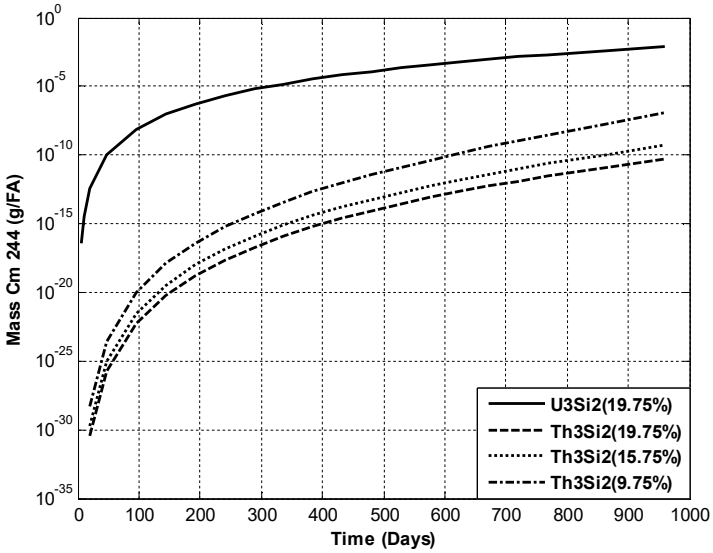

FIG. 5. The mass of Cm-244 depending on the time of operation.

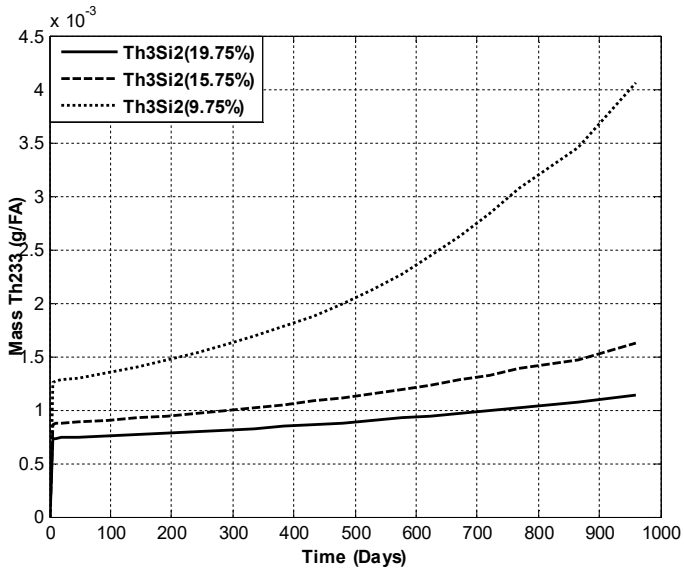

FIG. 6. The mass of Th-233 depending on the time of operation.

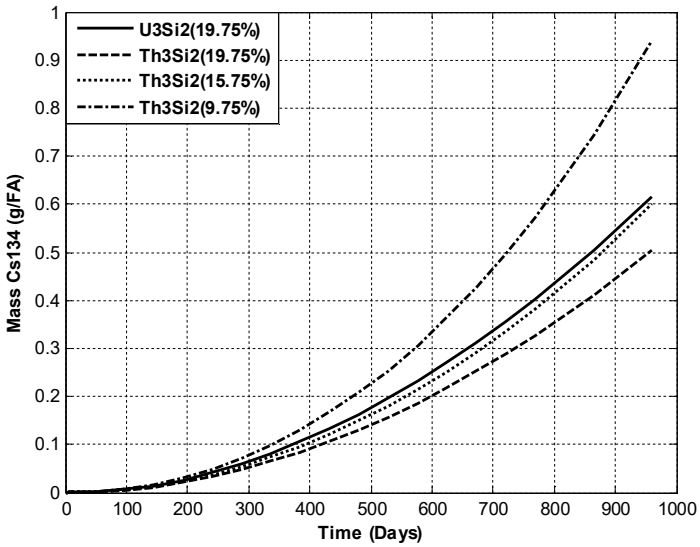

FIG. 7. The mass of Cs-134 depending on the time of operation.

First of all, we can observe that the accumulation of actinides (Pu-239 and Cm-244) in variant V_1 (standard variant) is much higher than in other variants (V_2, V_3 and V_4). Obviously, the reason for this big difference is the presence of uranium 238 in the standard variant.

On the other hand, we can see that, the accumulation of isotopes (Pu-239, Cm-244, Th-233 and Cs-134) in variant V_4 is bigger than in variants V_2 and V_3, and that is because of the higher neutron flux, as presented in Figs. 2.a and 2.b.

Storage of Spent Fuel in Cooling Pool

Figs. 8 and 9 show the total residual decay heat and activity of spent nuclear fuel (SNF) depending on the cooling time for all variants. The burnup of nuclear fuel was 100 MW.day/KgU.

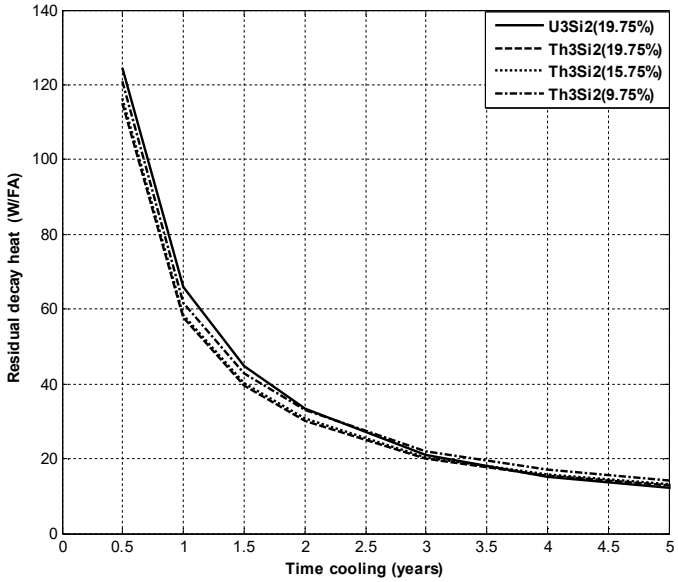

FIG. 8. Residual decay heat on SNF depending on the cooling time.

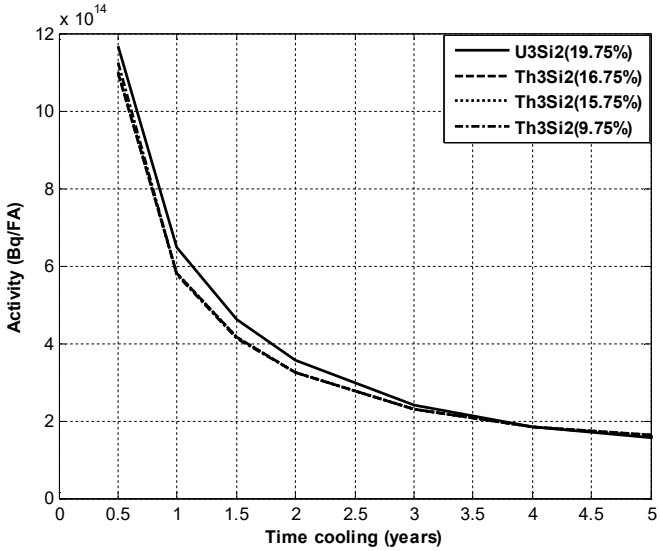

FIG. 9. Activity of SNF depending on the cooling time.

The residual energy and activity for variant V_1 (standard variant) are higher than in the other variants in the first 3 and 4 years, respectively. This result leads us to suggest that replacing uranium silicide with thorium silicide is safer at the beginning of the cooling time, which is the most energetic state of spent nuclear fuel.

The difference in residual energy and activity is due to the difference in the isotope concentrations accumulated as a result of fission, especially the fission product, since the nascent

isotopes with short half-lives, such as (^{134}Cs, ^{242}Cm ^{106}Ru and ^{144}Ce), accumulate more in uranium silicide than in thorium silicide, which explains the increase in residual energy and activity at the beginning of the cooling time.

As for isotopes with a greater half-life, such as (^{137}Cs and ^{90}Sr), they are more concentrated in depleted thorium silicide than in depleted uranium silicide, which explains the increase in residual heat and activity in spent thorium silicide after 4 years of cooling of the spent fuel.

Conclusion

In this paper, the effect of replacing uranium silicide with thorium silicide in the FA of the JRTR on the neutronics and radioactive properties of the fuel inside the plant is studied. The resulting calculations have shown that the replacement process leads to several positive aspects:

1) In V_1 and V_3, where the values of K_{inf} are equal, there is an increased neutron flux in thorium silicide, which means an increased radioiodine productivity during operation.
2) Increasing the length of the cycle time for fuel combustion when the enrichment rate is equal or reducing the enrichment to reach the same cycle duration, thus reducing the consumption of nuclear fuel, especially uranium, will be preferred as uranium will face in the near future a global shortage in its presence as a result of large global consumption [8].
3) The coefficient of thermal reactivity and the coefficient of uneven distribution of energy release for the FA containing thorium silicide are approximately equal to those of uranium silicide, with a slight preference for thorium silicide.
4) The residual energy in thorium silicide after its combustion and during the first cooling years is less than that of uranium silicide, which increases the safety margin in nuclear plants in case of accidents.

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