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Burn up Theoretical Analysis of A Thorium Fuel Rod in Light Water Reactor

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ABSTRACT

A computer model was designed to analyze the burn up and irradiation of both Th-Pu and Th-U fuel rod in a typical light water reactors conditions. MCNP computer model was designed to simulate the fuel rod burnup and evaluate neutron flux and group constants . A system of ordinary differential equations were solved numerically to evaluate the isotopic concentrations for both the two types of fuel using the previous calculated data from MCNP model. The results are analyzed and compared with published data where satisfactory agreement was found.

INTRODUCTION

The introduction of the Thorium – based nuclear fuel would therefore vastly enlarge the fissile resources by breeding ^{233}U. Thorium worldwide resources are three times larger than uranium resources. Thorium fuel has also another advantages: reduction in fuel cycle cost , reduction in ^{235}U enrichment requirements , safer reactor operation because of lower core excess reactivity requirements. Thorium fuel is irradiated to higher burn up in conditions similar to light water reactors. Two types of fuels are tested: 1- Thorium oxide (ThO_2) mixed with PuO_2 2- Thorium oxide (ThO_2) mixed with slightly enriched uranium oxide. The criticality and neutronic characteristic of the irradiated fuel are simulated using computer codes. ^{232}Th is a fertile material which changes to ^{233}U due to neutron absorption while other fissile isotopes such as ^{235}U or Pu isotopes deplete during fuel irradiation. ^{232}Th converts rapidly to ^{233}U such that the reactor depends on the fission of ^{233}U [1,2,3,4].

MCNP computer code, which is based on Monte Carlo method, are used to simulate the burn up and irradiation of fuel rod in a conditions similar to LWR reactor. Two types of fuel rod are considered: Th-Pu and Th-U. MCNP code determine the neutron flux , K_{eff} , and absorption and fission group constants[5,6,7]. A system of ordinary differential equations are solved numerically to evaluate the evolution and burn up of different isotopes with time. The results are compared with benchmark problem to validate both the model and the results.

MATHEMITICAL MODEL AND DATA

The following section describe both the material composition and dimensions of the two types of fuel used in the analysis

1- Thorium – Plutonium [Th-Pu] fuel rod

A fuel rod with radius 0.47 cm sourded by clad thickness 0.07 cm , the water coolant channel width is 0.31 cm [1] the composition of fuel , clad and coolant is given in the following table :

Table 1 Isotopic Composition of Th-Pu fuel rod [1]

	Isotopes	Concentration atom/barn.cm
Fuel Temperature = 1023 K Power 211 W/cm	Th-232 Pu-238 Pu-239 Pu-240 Pu-241 Pu-242 O	2.22E-2 9.72E-6 5.99E-4 2.32E-4 7.69E-5 4.78E-5 4.41E-2
Clad	Cr Fe Zr	8.14E-5 1.60E-4 4.37E-2
Coolant Temperature=583 K	Cr Mn Fe Ni C H O	3.2E-4 2.11E-5 8.46E-4 3.76E-4 2.68E-6 4.8E-2 2.4E-2

2- Thorium – Uranium [Th -U] Fuel Rod

Table 2 isotopic composition of Th-U fuel [8,9]

Zone	Isotopes	Concentration atom/barn.cm
Fuel Temperature = 1611 K Specific power =37.935 Kw/kg	U-235 U-238 Th-232 O	1.04062E-3 4.241746E-3 1.6167017E-2 4.28725E-2
Clad Temperature = 750 K	Cr Fe Zr	8.14E-5 1.60E-4 4.37E-2
Coolant Temperature=605 K Density = 0.644 g/cc	H O	4.3091E-2 2.1545E-2

MCNP computer code is used to model the geometry and compositions of Th-Pu and Th-U fuel rod with the data given in tables No. 1 and 2. The neutron flux Φ in the fuel rod is calculated. The energy group constant for both radiative capture (σ_γ) and fission (σ_f) cross sections are determined for every isotopes by calculating radiative capture and fission rate and dividing it by the neutron flux.

The neutron flux φ is calculated using MCNP code model[11] for each type of fuel and energy group g , and is given by the following relations

$$\phi_g = \frac{1}{V}\int_V \int_E \phi_{g,k}(E)dEdV \tag{1}$$

where the integration over E is carried out over energy boundaries and fuel zone

The capture and fission cross section for each isotope calculated from the relations

$$\sigma_g = \frac{\int_E \phi\ (r,E)\sigma(E)\ dE}{\int_E \ \phi(r,E)dE} \tag{2}$$

Φ neutron flux for fuel rods
The integration is carried out over fuel zone volume and all ranges energy groups.

Fuel Burnup:-

During reactor operation , Plutonium isotopes and ^{235}U are burned due to neutron absorption and fission and ^{233}U builds up due to neutron absorption of ^{232}Th. . More details for Thorium and Plutonium chains can be found at

The time rate of change of i isotopes is given by [10,12]]:

$$\frac{dN^i}{dt} = \lambda^j N^j + \sigma_c^j \phi N^j - \lambda^i\ N^i - \sigma_a^i\ \phi N^i \tag{3}$$

where :-

In [Th-Pu] fuel we calculates the depletion and buildup of the following isotopes ^{232}Th , ^{233}Th , ^{233}Pa , ^{233}U , ^{238}Pu ,^{239}Pu , ^{240}Pu , ^{241}Pu , ^{242}Pu , ^{243}Pu , ^{241}Am , ^{242}Am and ^{243}Am.

In [Th-U] fuel we follow the following isotopes:
^{235}U , ^{238}U , ^{232}Th , ^{233}Th , ^{233}Pa , ^{233}U , ^{238}Pu ,^{239}Pu , ^{240}Pu , ^{241}Pu , ^{242}Pu , ^{243}Pu , ^{241}Am , ^{242}Am and ^{243}Am.

.σ_c capture cross section for isotopes j which leads to formation of isotopes i.

.σ_a absorption cross section and λ decay constant for isotope i
φ neutron flux at each fuel element and energy group

RESULTS AND DISCUSSION

Table 3 Comparison between Cross section for Different Isotopes at Fresh fuel

Isotope	σ_f (present)	.σ_f (reference)	σ_a (present)	σ_a reference
Th-232	0.032653	0.029	0.722833	0.7889
Th-233	2.64038	--	50.87138	--
Pa-233	0.2038	0.1631	22.9186	24.95
U-233	34.3184	35.45	39.182	40.6
Pu-238	2.06754	1.981	19.56154	17.94
Pu-239	40.0659	44.63	60.4169	69.74
Pu-240	0.6963	0.63	41.114	49.72
Pu-241	51.3658	55.03	67.857	72.97
Pu-242	0.542	0.4502	19.4605	33.91
Pu-243	19.5237	--	28.95325	--
Am-241	0.94712	0.8569	56.17062	62.47
Am-242m	284.0056	289.3	339.8556	346.0
Am-243	0.56528	0.4479	45.78178	49.67
Cm-242	0.5411	0.4503	4.0341	5.699
Cm-243	68.782	72.51	76.6883	80.97
Cm-244	1.04862	0.9772	14.62532	18.29

Both Th-Pu and Th-U fuel rods are burned and irradiated in PWR conditions up to 60000 MWd/T. At Th-Pu fuel ^{233}U is produced and Plutonium isotopes are consumed. At Th-U fuel ^{233}U and Pu isotopes are produced while ^{235}U is consumed.

<u>Comparison with Published data</u>

Table 3 compare the fission and absorption cross section for different isotopes at fresh fuel . Equation 1 and 2 are used at the present model to evaluate the proposed cross section. The results indicate agreement in most cases and a difference appears at ^{239}Pu .

Figure 1 illustrates the ratio of Pu(t)/Pu(0) versus burn up (MWd/T) for Th-Pu fuel the results indicates that Pu isotopes are consumed at Th-Pu fuel the total Pu ratio is 0.47 at 30000 MWd/T and 0.238 at 60000 MWd/T , which means that burn up of Th-Pu fuel consumes the Pu isotopes with ratio 76.2 % at 60000 MWd/T. Good agreement was found between the present MCNP model and reference 1. The difference is 3 and 6 % at 30000 and 60000 MWd/T

Figure 2 illustrates the ratio between Pu-fissile/Pu(0) , versus burnup (MWd/T) for Th-Pu fuel . The results indicates that Pu- fissile isotopes (^{239}Pu , ^{241}Pu and ^{243}Pu) are consumed with burn up the ratio starts woth ratio 0.7 at fresh fuel and decreases to 0.49 and 0.38 at 30000 and 60000 MWd/T respectively . the difference between present MCNP model and the reference results increases with burnup at 60000 MWd/T MCNP value is 0.38 and the reference value is 0.24

Figure 3 illustrates the ratio of ^{233}U/initial fissile Pu. versus burnup (MWd/T) for Th-Pu isotopes. The results indicates that ^{233}U concentration increases with burnup . It approach to 0.39 and 0.47 at 30000 and 60000 MWd/T respectively. good agreement was observed between the present model and the reference results at all burn up values.

Burnup of Th-Pu Fuel

Figure 4 illustrates the behavior of Pu isotopes versus burn up (MWd/T) for Th-Pu fuel . The results indicates that ^{239}Pu , ^{240}Pu , and ^{241} Pu decrease with burn up. ^{242} Pu increase with burn up. ^{239}Pu concentration decreases from 5.99E-4 at fresh fuel (atom/barn.cm) to 1.2055E-4 and 2.52E-5 at 30000 and 60000 MWd/T respectively.

Figure 5 illustrates the behavior of Am isotopes versus burnup (MWd/T) for Th-Pu fuel Am isotopes increase with burnup but the concentration is small compared with Pu isotopes.

Figure 6 illustrates the production of ^{233}U isotope with burn up (MWd/T) fot Th-Pu fuel at 30000 MWd/T the concentration is 2.275E-4 and at 60000 MWd/T the concentration is 3.079E-4.

- Burnup of Th-U Fuel

Figure 7 illustrates the production of fissile plutonium isotopes (^{239}Pu , ^{241}Pu) and ^{233}U in Th-U fuel versus burnup (MWd/T) . The results indicates that ^{239}Pu approach asymptotic at 30000 MWd/T. ^{239}Pu builds up from 0 at fresh fuel up to concentrations 2.811E-5 , 3.6746E-5 , 3.91582E-5 and 3.9011E-5 (atom/barn.cm)at burnup 10000 , 20000, 30000, and 60000 MWd/T respectively. ^{233}U builds up from 0 at fresh fuel to 0.89862E-4 0.1569E-3 ,0.19763E-3 and 0.2441E-3 atom/barn.cm at burn up 10000 , 20000 , 30000 and 60000 MWd/T.

Figure 8 illustrates the burnup of ^{235}U at Th-U fuel versus burnup (MWd/T) . ^{235}U initial concentration is 0.104062E-2 (atom/barn.cm) .Its concentration approach a value of 0.706E-3 ,0.47931E-3 , 0.325298E-3 and 0.102704E-3 , at burnup 10000, 20000 ,30000, and 60000 MWd/T respectively. i. e. ^{235}U are burned at rate of 32.13 % , 53.939 % , 0.68.74 and 90.13 % at burnup 10000 , 20000 , 30000 and 60000 MWd/T.

Figure 9 illustrates the ratio of both ^{233}U and ^{239}Pu to initial ^{235}U versus burnup of Th-U fuel. The results indicate that ratio of ^{239}Pu builds up to 0.037 at burn up 30000 MWd/T and then approach asymptotic behavior. ^{233}U ratio increase with burnup at 30000 MWd/T the value equal 0.189917 , while at 60000 MWd/T the ratio is 0.234

CONCLUSION

- A computer model was designed to analyze the burn of both Th-Pu and Th-U fuel and determine the isotopic composition under irradiations.

- In Th-Pu fuel , fissile Pu isotopes (^{239}Pu , ^{241}Pu and ^{243} Pu) are consumed and burned with burn up 79.8% at 30,000 MWd/T and 95.8 % at 60,000 MWd/T.

- In Th- U fuel both ^{239}Pu and ^{233}U are produced , the ratio of ^{239}Pu to initial ^{235}U is 0.037 with asymptotic behavior at 30000 MWd/T , while ^{233}U ratio is 0.1899 at 30,000 MWd/T and increases up to 0.234 at 60,000 MWd/T

- Th-Pu fuel are used to consume and burn Pu isotopes, then mixed plutonium with thorium are used to reduce the amount plutonium and proliferation resistance.

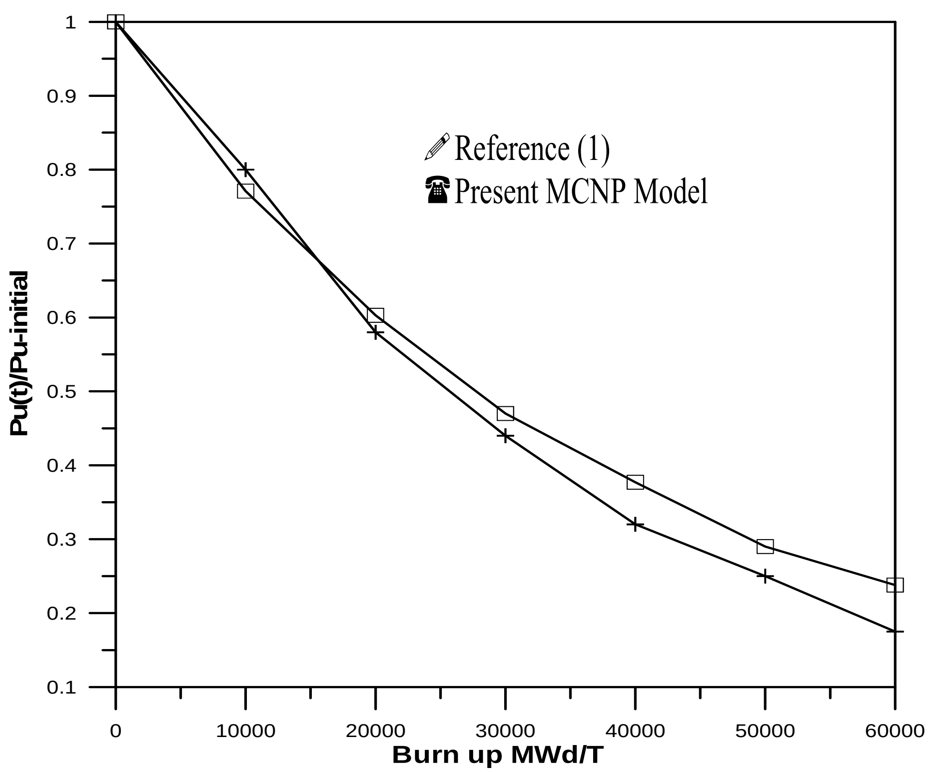

Figure 1 Ratio of total plutonium versus Fuel burnup (MWd/T)

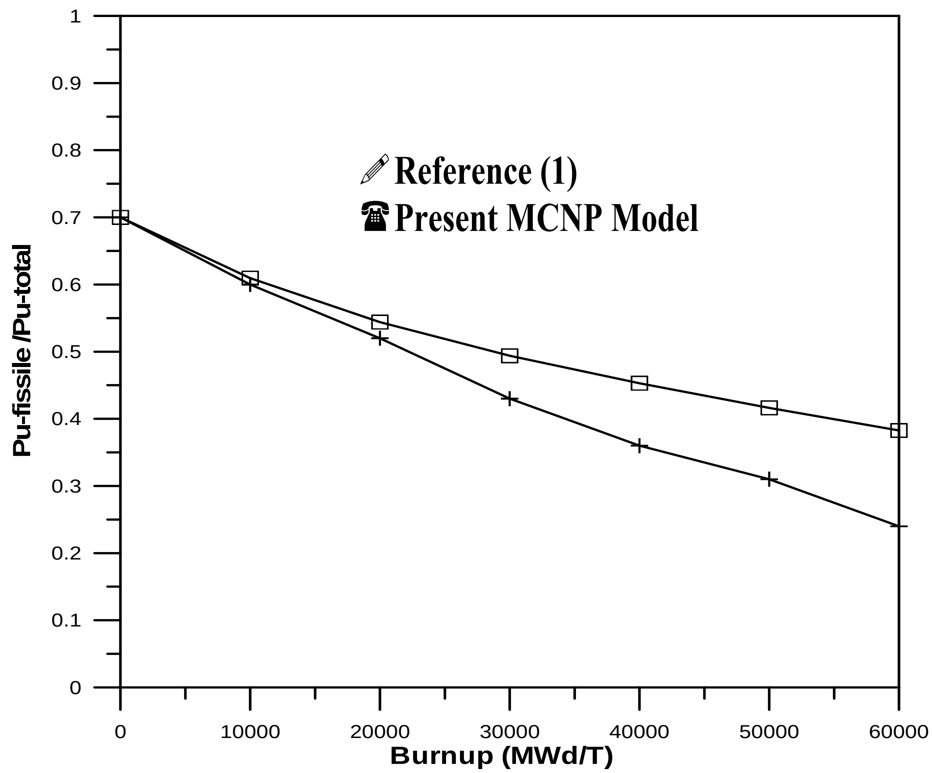

Figure 2 Ratio of fissile plutonium versus Burn up (MWd/T)

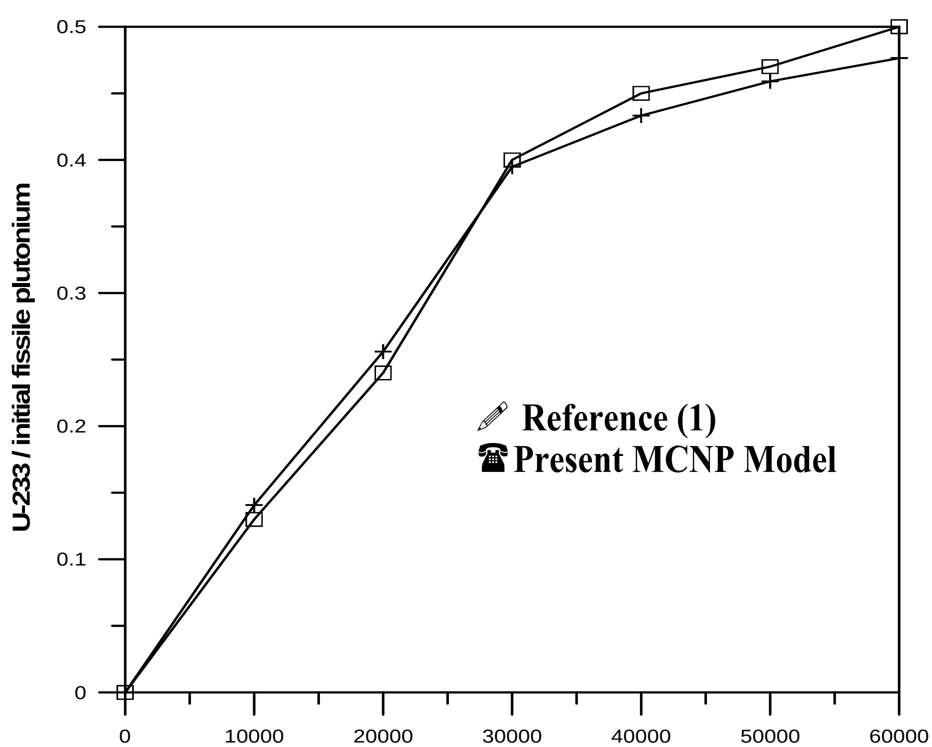

Figure 3 Ratio of ^{233}U to initial fissile plutonium versus burn up (Mwd/T)

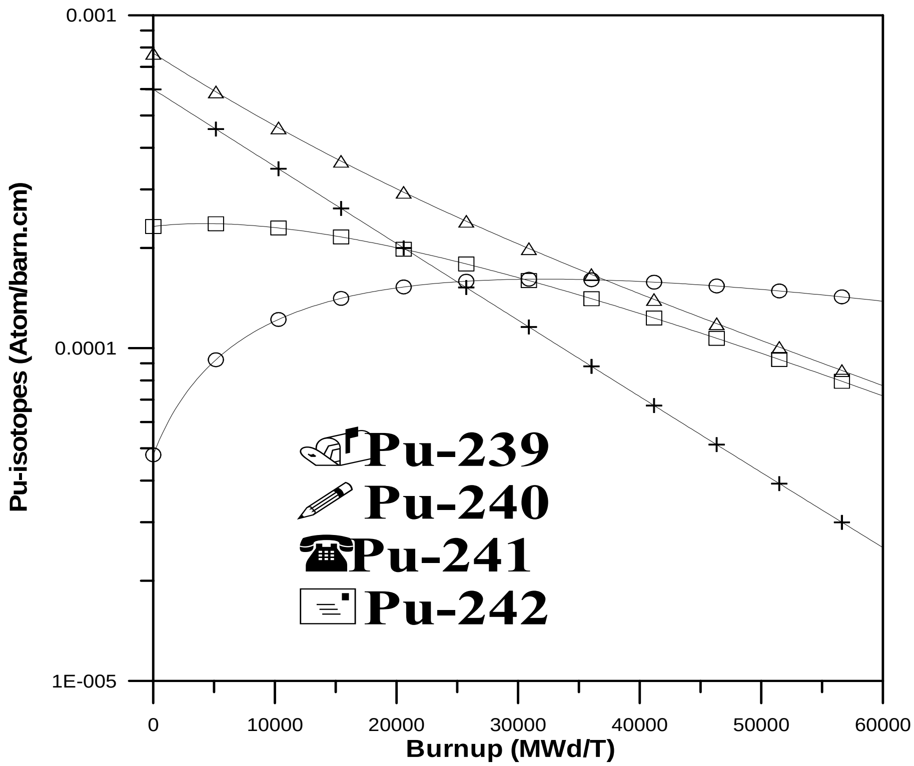

Figure 4 Burn up of Pu-isotopes in Th-Pu fuel versus burnup MWd/T

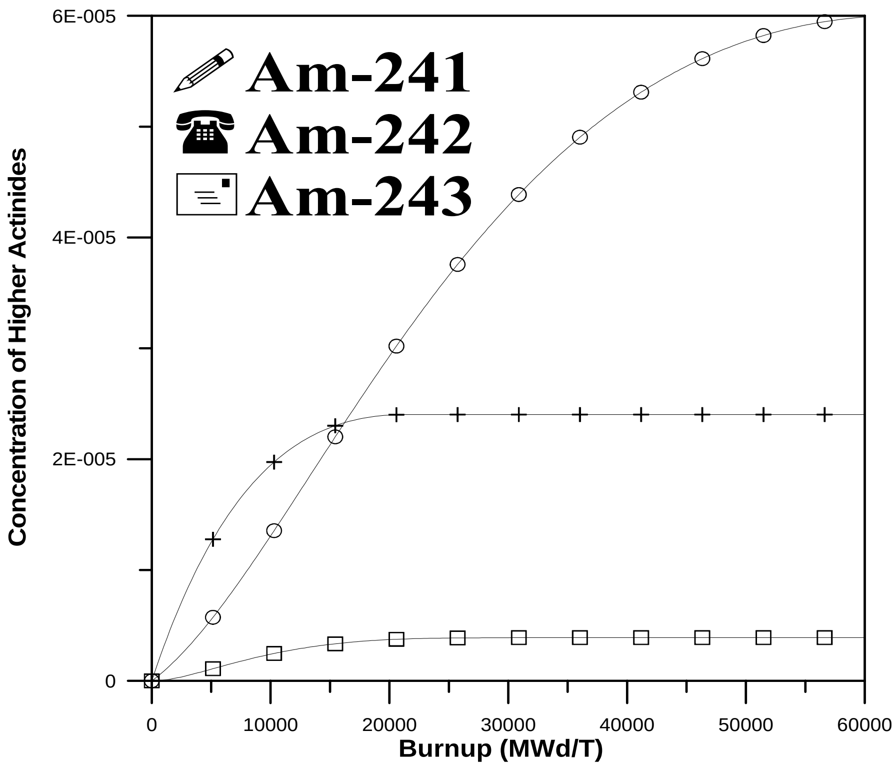

Figure 5 Concentration of Americium isotopes versus burnup (MWd/T) in Th-Pu fuel

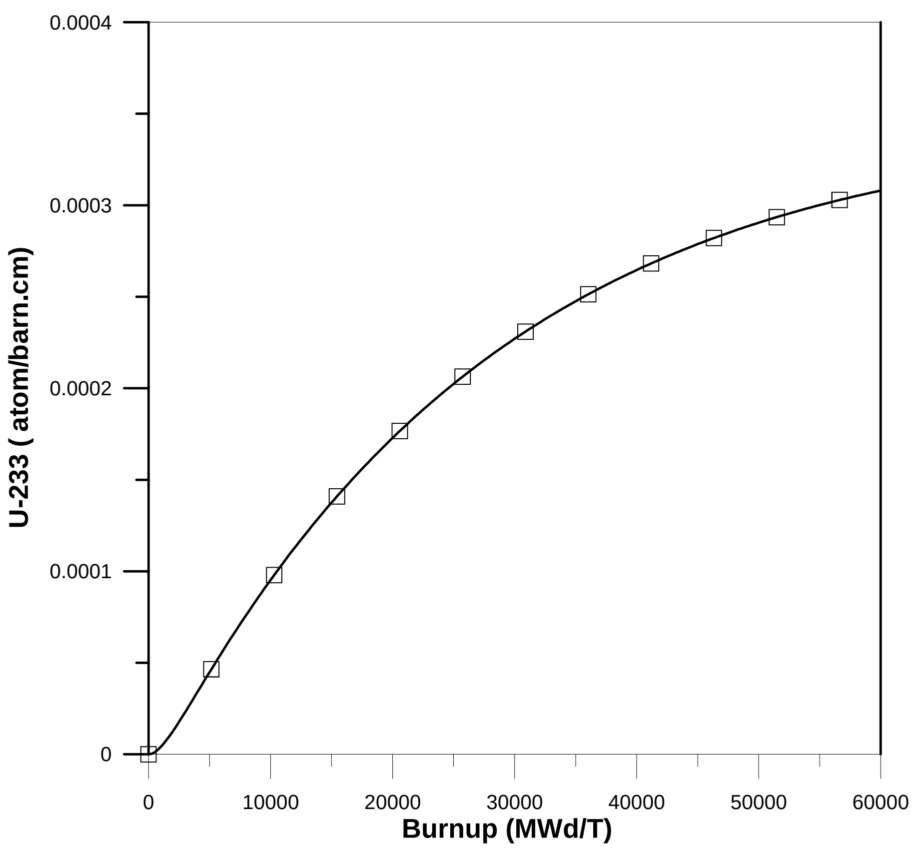

Figure 6 Production of U-233 versus burnup (MWd/T) in Th-Pu fuel

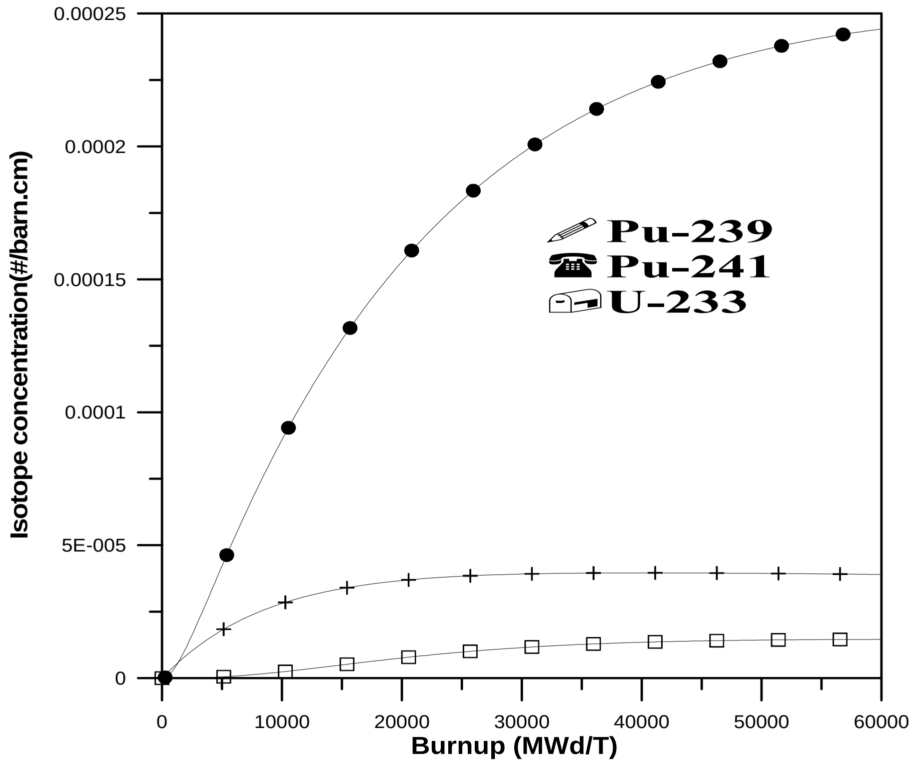

Figure 7 generation of fissile isotopes in Th-U fuel under burn up (MWD/T)

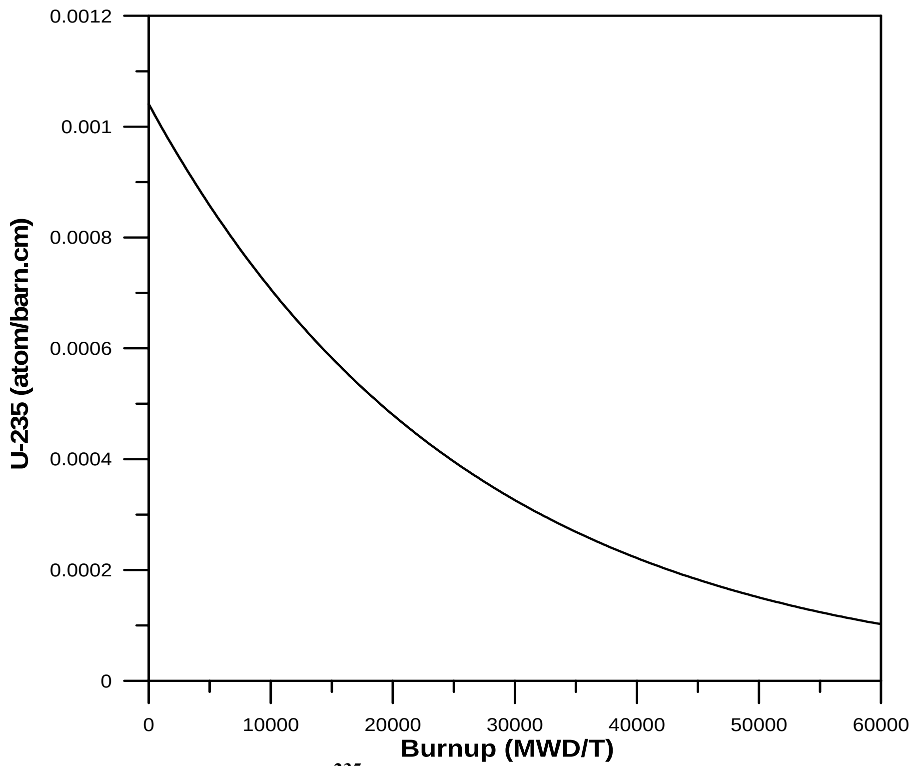

Figure 8 behaviour ^{235}U in Th-U fuel versus burnup MWd/T

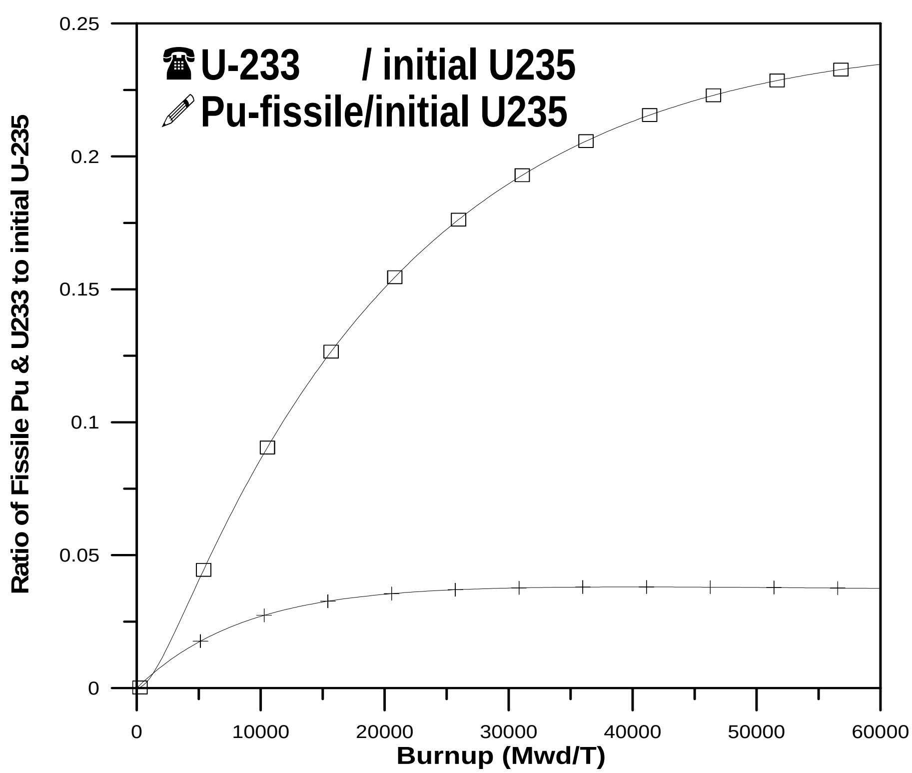

Figure 9 Ratio of Fissile plutonium and ^{233}U to initial ^{235}U versus Burnup (MWd/T) in Th-U fuel

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