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Original Article

Determination of plutonium and uranium content and burnup using six group delayed neutrons

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ABSTRACT

In this study, investigation of spent fuel was performed using six group delayed neutron parameters. Three used fuels (F1, F2, and F11) which are burnt over the years in the core of Missouri University of Science and Technology Reactor (MSTR), were investigated. F16 fresh fuel was used as plutonium free fuel element and compared with irradiated used fuels to develop burnup and Pu discrimination method. The fast fission factor of the MSTR was calculated to be 1.071 which was used for burnup calculations. Burnup values of F2 and F11 fuel elements were estimated to be 1.98 g and 2.7 g, respectively. ^{239}Pu conversion was calculated to be 0.36 g and 0.50 g for F2 and F11 elements, respectively.

1. Introduction

The nondestructive analysis (NDA) is the most common and a well-utilized technique which can be performed by either gamma spectroscopy or delayed neutron measurement for burnup and cooling time determination of spent fuel [1,2]. This technique is also used for discriminating mixed oxide (MOX) fuel from uranium dioxide (UO_2) fuel using gamma spectroscopy and multivariate statistical analysis [3–6]. The only disadvantage about gamma spectroscopy NDA method for discriminating MOX fuel from UO_2 fuel is that the method depends on the lowest energy peak measurements directly from Pu and U. These low energy peaks can be considered as background radiation since Pu and U emit low energy gamma rays which make Pu and U discrimination difficult in MOX fuel. Willman and coworkers overcame this problem and were able to discriminate the MOX fuel from UO_2 fuel by calculating the ratio of ^{134}Cs/^{154}Eu [3] while Dennis and Usman [1] proposed using ^{106}Ru peak ratios with cesium to discriminate between Uranium and Plutonium.

Another NDA method for fuel interrogations is to implement delay neutron measurements. Jordan and Perret were able to perform a delay thermal neutron measurement to investigate fresh and spent fuel at Paul Scherrer Institut in Switzerland. Akyurek and Usman [7] reported a new experimental technique to determine burnup and Pu content of spent fuel using delayed fast neutron energy spectroscopy at Missouri University of Science and Technology Reactor (MSTR). In this technique, burnup and Pu conversion values were obtained based on delayed fast neutron emission ratios for spent and fresh fuels at different power levels [7]. The main objective of this manuscript is to determine burnup and Pu conversion values of MSTR fuel elements using six group delayed neutron parameters and experimental delayed fast neutron ratios of different fuel elements (F1, F2, F11, and F16). MSTR was used to collect delayed fast neutron energy spectrum of four different fuel elements including F16 fuel element as a fresh fuel. Then, the fuel element neutron emission ratios were obtained within the specific time period [7] and these ratios were used in this study for theoretical burnup calculations using six group delayed neutron parameters.

MSTR is a pool-type research reactor which operates at 200 kW thermal power. The reactor core contains 4 controls rods and 15 fuel elements which are in the form of U_3Si_2Al with 19.75% enriched ^{235}U. A fresh fuel element contained 112.5 g of ^{235}U and 457.2 g of ^{238}U. Fig. 1 shows the current fuel configuration of the core. This core configuration is called T121 where “T” specifies “thermal” which means the core is pushed close to the graphite thermal column. The low enriched fuel elements have been

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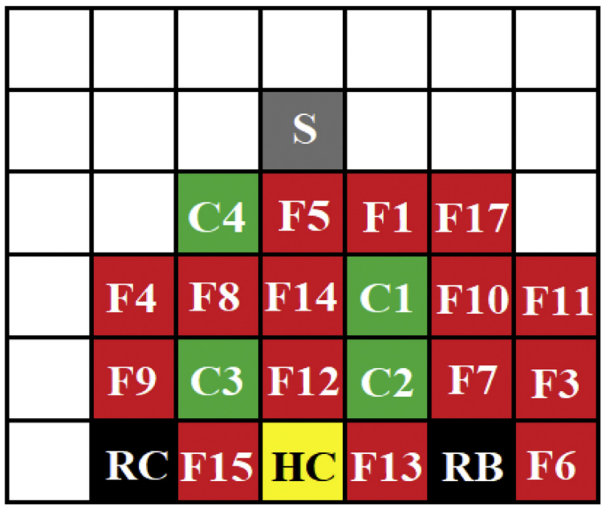

Fig. 1. Core configuration of MSTR. F and beside its number represent an individual fuel element, C represent a control rod, HC is the hot cell, RC is the cadmium rabbit tube, RB is the bare tube, S is the source for reactor startup. Rest of the white box represent blank plate in the core grid.

shuffled several times during the core life but burnup assigned to each fuel element is not accurate in that no credit is given to the actual location of the element during the operation. A single value of burnup is assigned to the entire core which is divided between the fuel elements present during the operation.

This almost random assignment does not specifically define the true burnup value of a fuel element. Therefore, correct burnup calculation for individual fuel elements is required for the reactor core. Present work offers one alternative approach for obtaining burnup credit and Pu conversion values correctly for Missouri's MSTR. Same approach can be applied to other materials testing reactors (MTR) to obtain burnup values for various fuel elements.

2. Material and method

Delayed neutrons are important for controlling nuclear reactors even though their production from the fission process is only approximately 1%. Each fission produces some prompt neutrons and fission fragments. Several of these fission fragments are neutron rich and decay by neutron emission which is the primary source of delayed neutrons in a fission reactor. In addition, successive neutron absorptions of ^{238}U in nuclear fuel produce ^{244}Cm and its spontaneous fission provide delayed neutron. There are approximately 250 well-known delayed neutron precursors. These precursors are generally represented by 6-groups although other studies have recommended 8-groups and 12-groups of delayed neutron based on half-lives of their precursors [8,9]. Using 6-group delayed neutron parameters (Table 1), the number of total neutrons after reactor shutdown can be expressed by Eq. (1), assuming all the delayed neutrons are released from decay of a fission product;

$$N(t) = \sum_i w_i . e^{-\lambda_i t} \tag{1}$$

where t is the time post shutdown and w_i is the fission yield of i'th precursor group (again the grouping is based on the half-life). We can agree on the use of six groups (i); a choice which is arbitrary yet very commonly used by the reactor analysis community. Each of these groups can be defined by one decay constant (λ_i), and one fractional yield (β_i). The term β_i is the fraction of all the fission neutrons emitted in fission that appear as delayed neutrons in the ith group. Table 1 shows information of 6-group delayed neutron parameters for thermal fission of ^{235}U, ^{233}U, and ^{239}Pu and fast fission of ^{238}U fissions;

$$N(t) = \left(A.\sum_i w_{i-U235}.e^{-\lambda_i t}\right)_{\substack{235\\thermal}U} + \left(B.\sum_i w_{i-U238}.e^{-\lambda_i t}\right)_{\substack{238\\fast}U} + \left(C.\sum_i w_{i-Pu239}.e^{-\lambda_i t}\right)_{\substack{239\\thermal}Pu} \tag{2}$$

Eq. (1) provides the time dependence of delay neutrons per fission. In nuclear reactors, majority of fissions take place by thermal neutrons and only small fraction of fissions are induced by fast neutrons. Fast fissions are considered in thermal reactors by defining a fast fission factor, commonly represented by ε. The fast fission factor is defined by the ratio of total number of fission neutrons (fast and thermal) to the number of fission neutrons from just thermal fissions [10]. The value of ε varies from 1.02 to 1.08 depending on the type of reactor [11]. Even for a given reactor fast fission factor can change due to the operating conditions. For example, at lower temperature a test reactor will have a higher thermalization and a lower fast fission factor than at higher pool water temperature. Calculation of the fast fission factor by various methods is available in the literature [12,13]. Arkani et al. [12] calculated the fast fission factor of 1.074 using Monte Carlo method for a MTR (Materials Test Reactor) research reactor.

For this study, fast fission factor value had to be calculated for MSTR fuel. One source that provided this information was Arkani et al. [12] which reported a value of 1.074. The second source of information was ALQahtani and Alajo [14] which provided the results of their SANDII simulation giving thermal neutron flux of 4.47E+11 n/cm^{2}s and the total neutron flux of 4.79E+11 n/cm^{2}s. Simply taking the ratio of the two value would provide the fast fission factor of 1.071 which is in very good agreement with Monte Carlo results published by Arkani et al. [12] for a similar reactor design.

In order to obtain full burnup calculations using six group delayed neutron parameters for a specific fuel element of MSTR core, Eq. (1) requires a correction by adding the contribution of thermal and fast fissions;where A represents the percentage of thermal fission of ^{235}U, B represents the percentage of fast fission of ^{238}U, and C represents the percentage of ^{239}Pu content. Eq. (2) gives the number of total delayed neutrons after reactor shutdown assuming all the delayed neutrons are released from decay of a fission product of ^{235}U, ^{238}U, and ^{239}Pu. If the fuel element is a fresh (new) fuel, it is assumed that the contribution of C (percentage) in Eq. (2) to the total number of delayed neutrons is zero. Therefore, total delay neutrons can be calculated from the first two terms of Eq. (2) for a fresh fuel.

In this study, three used (old) fuels (F1, F2, and F11) which are burnt over the years in the core of MSTR were investigated. The reactor fuel elements are shuffled at certain time intervals and some used/fresh fuel elements are protected in the pool storage where they are separated from reactor core. Fresh fuel element F16 remained in the reactor pool storage for 26 years before we took delayed neutron emission rates of these fuel elements (F1, F2, F11, F16) for this work. F16 fuel can be used as plutonium free fuel element and compared with irradiated used fuels to develop burnup and Pu discrimination method.

Based on the published and agreed upon delayed neutron fractions for the six delayed-neutron group and their respective half-lives (as shown in Table 1) neutron emission rate from thermal fission of ^{235}U, and ^{239}Pu were calculated as a function of post

Table 1
Table of six group delayed neutron parameters [15].

Group	Decay Constant λ_i (1/s)	Mean Life $\tau = 1/\lambda_i$ (sec)	Half Life $\ln 2/\lambda_i$ (sec)	Fission Yield w_i (n/fission)	Fraction Yield β_i
Thermal Fission in ^{235}U					
1	0.0124	80.645	55.720	0.00052	0.000215
2	0.0305	32.786	22.720	0.00346	0.001424
3	0.111	9.009	6.220	0.0031	0.001274
4	0.301	3.322	2.300	0.00624	0.002568
5	1.14	0.877	0.610	0.00182	0.000748
6	3.01	0.332	0.230	0.00066	0.000273
Total				**0.0158**	**0.0065**
Thermal Fission in ^{233}U					
1	0.0126	79.365	55.000	0.00057	0.000224
2	0.0337	29.673	20.570	0.00197	0.000777
3	0.139	7.194	5.000	0.00166	0.000655
4	0.325	3.076	2.130	0.00184	0.000723
5	1.13	0.884	0.615	0.00034	0.000133
6	2.5	0.44	0.277	0.00022	0.000088
Total				**0.0066**	**0.0026**
Thermal Fission in ^{239}Pu					
1	0.0128	78.125	54.280	0.00021	0.000073
2	0.0301	33.222	23.040	0.00182	0.000626
3	0.124	8.064	5.600	0.00129	0.000443
4	0.325	3.076	2.130	0.00199	0.000685
5	1.12	0.892	0.618	0.00052	0.000181
6	2.69	0.371	0.257	0.00027	0.000092
Total				**0.0061**	**0.0021**
Fast Fission in ^{238}U					
1	0.0132	75.757	52.380	0.00054	0.00019
2	0.0321	31.152	21.580	0.00564	0.00203
3	0.139	7.194	5.000	0.00667	0.0024
4	0.358	2.793	1.93	0.01599	0.00574
5	1.41	0.709	0.490	0.00927	0.00333
6	4.02	0.248	0.172	0.00309	0.00111
Total				**0.0412**	**0.0148**

shutdown time. Similarly, for ^{238}U fast fission based delayed neutron were calculated using the delayed neutron fraction data (Table 1) for fast fission. Knowing the fuel enrichment of the original fuel (19.75% enriched in ^{235}U and 80.25% ^{238}U) and MSTR fast fission factor of 1.071 delayed neutron emission rate as a function of time was generated using Eq. (2). In our previous work [7], we found that neutron emission rate between 150 and 400 s was a reliable indicator of burnup. Using the data from Akyurek and Usman [7] for the ratio of fresh fuel (fuel element 16) and other fuel elements (F16/F1, F16/F2 and F16/F11) after 100 kW run was obtained as shown in Table 2. Burnup of these elements were previously calculated using the experimental data as reported by Akyurek and Usman [7]. In order to interrogate nuclear fuel elements and determinate the Pu content, non-destructive analysis (NDA) was performed at MSTR using N-Probe neutron detector [16]. The detection system was placed in front of the beam port which is in the basement of MSTR. The F11 fuel shown in Fig. (1) is the nearest fuel to the beam port and sees the detector directly through the beam port in the case of W mode fuel configuration. Fuel elements were irradiated for 15 min at 100 kW power and the reactor was shut down. The delayed neutron measurements were taken right after shutdown for 30 min. The same data collection was repeated with other fuel element (F1, F2, F11, and F16) at the same location in the reactor core. Using the delayed neutron data one can predict burnup and compare it with the experimental data.

Table 2
Delayed neutron emission ratios and corresponding errors at 100 kW power [7].

Fuel fractions	Average	Uncertainty
F16/F1	1.1687	±0.0406
F16/F2	1.0180	±0.0776
F16/F11	1.0245	±0.0827

The prediction was the most important aspect of taking the experimental delayed neutron ratios as this information is unknown how much of the fuels have fissioned and produced plutonium. Therefore, if neutron emission rates of fuels are known, theoretically the delayed neutron numbers of other fuels would be revealed.

Table 2 shows the average delayed neutron ratios for indicated fuels (F1, F2, F11, and F16) after 100 kW power run. Experimental fuel ratios required for the theoretical burnup calculations of fuels were taken between 150 and 400 s after a 100 kW run. Akyurek and Usman explained why these time intervals were taken into account in their previous work at 100 kW reactor powers [7]. The delayed neutrons produced at low powers decay away faster than high power leading to poor statistics. Therefore, high level power such as 100 kW power were desirable for the delayed neutron analysis. Jordan and Perret similarly have chosen the time interval between 60 and 240 s for their fuel analysis with thermal neutrons [2]. Since F16 is assumed to be a new fuel element, it is expected that the fuel fractions (F16/F1, F16/F11 or F16/F2) should be greater than 1.0. Due to the statistical nature of the data some values did not produce a ratio of greater one but the average values were higher than 1.0. While poor statistics in the data is recognized, in the absence of better data all calculations are based on the average values shown in Table 2. For example, the average fuel fraction of F16 and F1 in this table is 1.1687 and its uncertainty is ±0.0406 which is ±4.06%.

Once the experimental data on the ratio of the neutron emission rate from F16/F1 (for example) is obtained and using the theoretical delayed neutron emission rate from the fresh fuel element (F16) one can calculate the emission rate from F1. The difference between the two can be attributed to the burnup. Using this value and the original amount of 112.5 g of ^{235}U in the fuel one can easily calculate the amount of burnup for the fuel element. This process was repeated for the other fuel elements.

Table 3
MSTR research reactor core fuel content since 1992.

Total kW/hr to date	Fuel Content (gram)			Fuel weight percentage		
	^{235}U	^{238}U	^{239}Pu	^{235}U (%)	^{238}U (%)	^{239}Pu (%)
0.00	1687.50	6856.80	0.0000	0.1975	0.8025	0.0000
3556.98	1687.34	6856.77	0.0285	0.1975	0.8025	0.0000
12205.47	1686.96	6856.70	0.0977	0.1975	0.8025	0.0000
16793.29	1686.76	6856.66	0.1344	0.1974	0.8026	0.0000
25707.64	1686.38	6856.59	0.2057	0.1974	0.8026	0.0000
31450.79	1686.13	6856.55	0.2517	0.1974	0.8026	0.0000
38239.72	1685.83	6856.49	0.3060	0.1973	0.8026	0.0000
48390.32	1685.39	6856.41	0.3872	0.1973	0.8027	0.0000
55710.84	1685.07	6856.36	0.4358	0.1973	0.8027	0.0001
61842.82	1684.80	6856.31	0.4849	0.1972	0.8027	0.0001
65676.15	1684.64	6856.28	0.5157	0.1972	0.8027	0.0001
78537.73	1684.07	6856.18	0.6186	0.1972	0.8027	0.0001
112199.37	1682.61	6855.91	0.8879	0.1970	0.8029	0.0001
142557.52	1681.28	6855.67	1.1282	0.1969	0.8030	0.0001
183354.97	1679.50	6855.35	1.4512	0.1967	0.8031	0.0002
202098.60	1678.68	6855.20	1.5996	0.1967	0.8031	0.0002
216161.89	1678.06	6855.09	1.7109	0.1966	0.8032	0.0002
226442.88	1677.61	6855.01	1.7923	0.1966	0.8032	0.0002
244391.91	1676.83	6854.86	1.9344	0.1965	0.8033	0.0002
256130.69	1676.31	6854.77	2.0273	0.1964	0.8033	0.0002
282811.08	1675.20	6854.57	2.2292	0.1963	0.8034	0.0003
305806.49	1674.19	6854.39	2.4112	0.1962	0.8035	0.0003
341337.99	1672.64	6854.11	2.6925	0.1961	0.8036	0.0003

3. Results

Fuel elements in the reactor core have been shuffled over time and continued to burn since 1992. In this study, we calculated the burnup value and Pu conversion of the F1, F2, and F11 spent fuels based on the fuel element F16, which was assumed to be fresh fuel, using the NDA method. Table 3 shows fuel content and weight percentage calculations of the ^{235}U, ^{238}U, and ^{239}Pu isotopes from MSTR log book information based on the total power run over the years. The fuel content and weight percentages given in Table 3 contain the information of all fuels actually present in the reactor core. For all fuel elements in the reactor core, the amount of burnt ^{235}U and ^{238}U are 14.86 g and 2.69 g, respectively. On the other hand, the amount of ^{239}Pu formed in the core of MSTR is 2.69 g.

As discussed earlier, total delay neutrons for a fresh fuel (F16) can be calculated with respect to time from first two terms of Eq. (2). The fast fission factor of the MSTR is calculated to be 1.071 which shows 6.63% is the fast fission and 93.37% is the thermal fission of all fissions. The constants A and B in Eq. (2) are 0.9337 and 0.0663, respectively. The delayed neutron numbers can be calculated using experimental delay neutron emission ratios for used fuel elements (F1, F2, and F11). For example; to find the delayed neutron numbers with respect to time for F2 fuel, we used F16/F2 ratio (1.0180) from Table 2. Then, average delayed neutron numbers of F2 fuel element obtained using this ratio for 100 kW power. The delayed neutron numbers difference between F16 and F2 fuel elements will actually provide the burnup of F2 element. Table 4 shows the delayed neutron numbers of F16 and F2 fuel elements between 150 s and 170 s.

Fig. (2) shows delayed neutron numbers of F1, F2, F11, and F16

Table 4
Theoretical delayed neutron emission rates and burnup with fast fission information.

Time (sec)	U235 Thermal (100%)	U238 Fast (100%)	Pu239 (100%)	Fuel 16 with fast fission factor (#n)	Fuel 2 with fast fission factor (#n)	F16 − F2 difference (#n)	Fuel 2 Burnup (g)
150	0.000116610	0.000120282	5.07048E-05	2.79027E-05	2.74106E-05	4.92072E-07	1.98
151	0.000114541	0.00011786	4.97227E-05	2.73923E-05	2.69093E-05	4.83072E-07	1.98
152	0.000112517	0.000115496	4.8763E-05	2.68933E-05	2.6419E-05	4.74271E-07	1.98
153	0.000110536	0.000113189	4.78253E-05	2.64053E-05	2.59396E-05	4.65664E-07	1.98
154	0.000108597	0.000110938	4.69089E-05	2.5928E-05	2.54707E-05	4.57247E-07	1.98
155	0.000106699	0.00010874	4.60133E-05	2.54611E-05	2.50121E-05	4.49014E-07	1.98
156	0.000104842	0.000106595	4.51379E-05	2.50045E-05	2.45635E-05	4.40961E-07	1.98
157	0.000103024	0.0001045	4.42823E-05	2.45578E-05	2.41247E-05	4.33083E-07	1.98
158	0.000101244	0.000102455	4.34459E-05	2.41207E-05	2.36954E-05	4.25376E-07	1.98
159	9.95014E-05	0.000100458	4.26282E-05	2.36931E-05	2.32753E-05	4.17835E-07	1.98
160	9.77951E-05	9.85075E-05	4.18289E-05	2.32747E-05	2.28643E-05	4.10457E-07	1.98
161	9.61243E-05	9.66028E-05	4.10473E-05	2.28653E-05	2.24621E-05	4.03236E-07	1.98
162	9.4488E-05	9.47426E-05	4.0283E-05	2.24646E-05	2.20684E-05	3.96169E-07	1.98
163	9.28854E-05	9.29254E-05	3.95356E-05	2.20724E-05	2.16831E-05	3.89253E-07	1.98
164	9.13158E-05	9.11502E-05	3.88048E-05	2.16885E-05	2.1306E-05	3.82483E-07	1.98
165	8.97783E-05	8.94159E-05	3.80899E-05	2.13127E-05	2.09368E-05	3.75856E-07	1.98
166	8.82721E-05	8.77214E-05	3.73908E-05	2.09448E-05	2.05754E-05	3.69367E-07	1.98
167	8.67965E-05	8.60656E-05	3.67069E-05	2.05846E-05	2.02216E-05	3.63015E-07	1.98
168	8.53507E-05	8.44474E-05	3.60378E-05	2.02319E-05	1.98751E-05	3.56795E-07	1.98
169	8.3934E-05	8.2866E-05	3.53833E-05	1.98865E-05	1.95358E-05	3.50704E-07	1.98
170	8.25458E-05	8.13202E-05	3.47429E-05	1.95483E-05	1.92035E-05	3.44739E-07	1.98

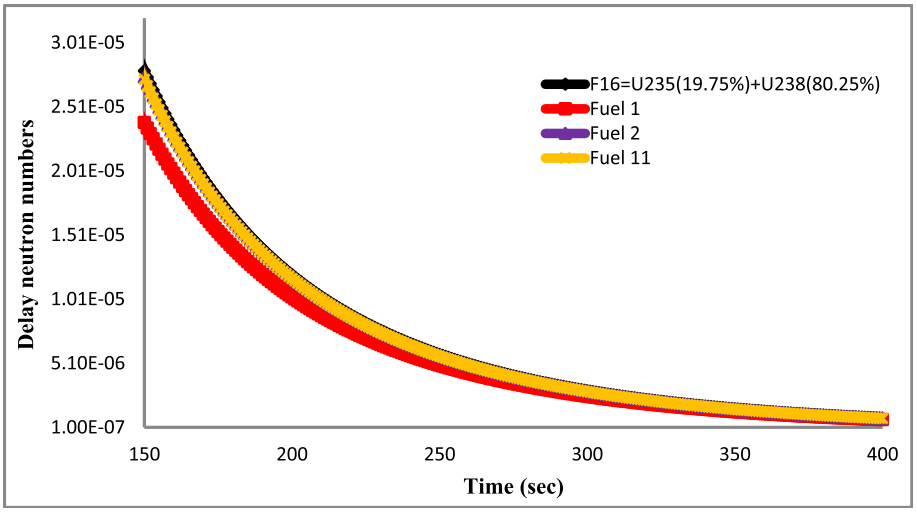

Fig. 2. Delayed neutron numbers with respect time at 100 kW power.

fuel elements with respect to time between 150 and 400 s for 100 kW. It should be noted that the number of neutrons with respect to time is obtained by Eq. (2) using the 6-group delayed neutron parameters. To get a better view of the difference in fuel, the figure was drawn closer and rescaled as shown Fig. (3). As shown in Fig. (3), it is understood that F16 fuel element, which is regarded as a new fuel, has produced higher delayed neutron emission than the other fuels. In fact the decay rate curves of delayed neutrons are in the order of burnups.

Table 5 shows the results of burnup and buildup data for the four fuel elements analysed. Burnup values of F2 and F11 fuel elements were estimated to be 1.98 g and 2.7 g, respectively. Their corresponding burnup values are 220 and 300.95 in MWD/T unit. Each fuel elements contain 112.5 g of ^{235}U. In order to find the burnup value of any fuel element, the theoretical emission rate of the F16 fuel element was obtained using six group delayed neutron parameters given in Eq. (2). As discussed earlier, the constants A, B, and C in Eq. (2) were found to be 0.0663, 0.9337, 0 for F16 fuel element, respectively. After theoretically calculating the time-dependent emission rate of the F16 fuel element, the experimental fuel ratios were taken from Table 2 and the theoretical delayed neutron emission rates of other fuels were calculated. In Table 2, three sets of neutron emission ratios and their corresponding uncertainty values are presented. Fuel element F1 has a sufficient burnup value which consequently produces smaller uncertainty. Unfortunately for the other two fuel elements the burnup is not sufficient and hence the calculation of uncertainty may overshadow the results but same technique can provide good quality results for ratios of distinctly different burnup elements. The delayed neutron emission difference between fresh fuel element (F16) and old fuel will provide the burnup value of old fuel elements (F1, F2, and F11). For example; the delayed neutron emission rate for F16 at 150 s is 2.79027E-05. Using the ratio of F16/F2 (which is1.0180) the delayed neutron number for F2 can be calculated as 2.74106E-05. Then,

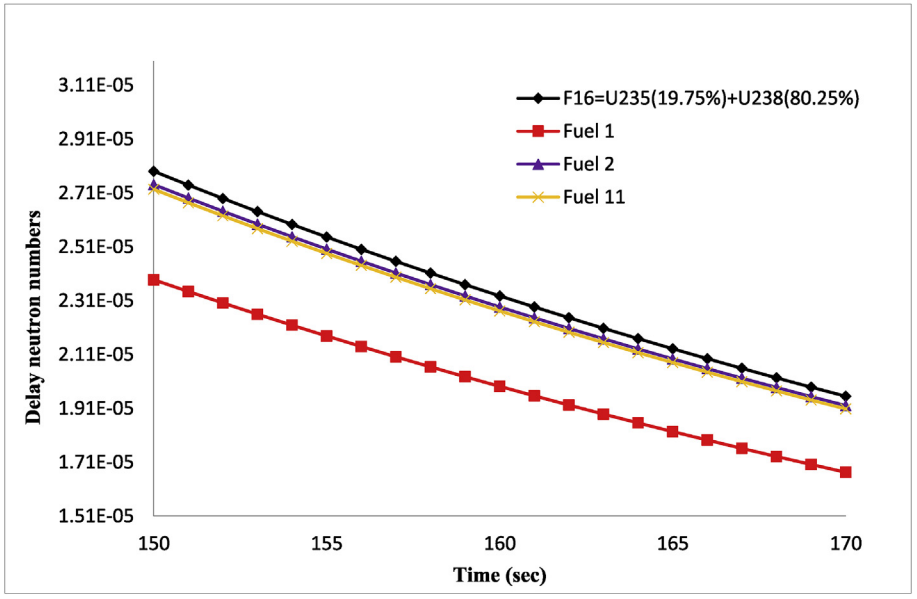

Fig. 3. Rescaled delayed neutron numbers with respect time between 150 and 170 s.

Table 5
MSTR research reactor fuel burnup and buildup data.

Fuel Elements	Burnup (g)	Remaining Fuel (g)	Burnup (MW-d/T)	^{235}U Burnup & Conversion	Pu conversion (g)
F16	0.00	112.50	0	0.00	0.00
F1	16.24	96.26	1810	19.22	2.99
F2	1.98	110.52	220	2.35	0.36
F11	2.70	109.80	300.95	3.19	0.50
Total	20.92	429.08	2330.95	24.76	3.85

F16 - F2 = 2.79027E-05 - 2.74106E-05 = 4.92072E-07 neutrons

Remaining ^{235}U fuel (%) = (2.74106E-05*100)/ 2.79027E-05 = 98.24%

Burnup = 112.5 - (Remaining ^{235}U fuel (%) * 112.5/100)

Burnup = 112.5 - (98.24*112.5/100 = 1.98 g for F2 fuel element

Using a conversion factor (1.1839) from the MS&T reactor, ^{239}Pu conversion calculated to be 0.36 g and 0.50 g for F2 and F11 elements, respectively. Total ^{239}Pu conversion for all three fuel elements was estimated as 3.85 g and this prediction is within the range of values reported to NRC. It should be noted that same value of Pu conversion can be calculated directly from the delayed neutron emission ratios for validation purposes.

In this study, the ^{239}Pu conversion obtained as 2.99 g for F1 fuel element using six group parameters and fast fission factor of MSTR. However, the ^{239}Pu conversion was calculated as 2.93 g for F1 fuel element using experimental method [7]. These two values are in reasonable agreement. Although F16/F1 = 1.1687 value is from experimental results [7] from 150 s to 400 s, the delayed neutron emission rate of F16 was obtained from Eq. (2). Using the delayed neutron emission rate of F16 and F16/F1 value from experimental work [7] at 100 kW power, the delayed neutron emission rate of F1 was calculated. The amount of plutonium conversion obtained by both using six group parameter study and experimental study [7] may be slightly different due to these theoretical and numerical calculations. Beside, F16/F1 value considered as 1.169 since it was recommended to use small significant digits (round off) even when the ratio of F16/F1 was 1.1687 in the previous study [7]. This may also have effected the result shifted a little. The third reason for this difference can possibly be explicit treatment of fast fission factor in the current study (Use of second term in Eq. (2)). In the previous study [7] B was set to zero and there was an overestimate for A and consequently an underestimate for C.

4. Conclusion

Based on the two independent sources [12,14] fast fission factor for MTR reactor is approximately 1.07 which is on a higher side than a typical PWR as one would expect. The assumption for the calculation of delayed neutron for fresh fuel was that fuel was enriched to 19.75% ^{235}U with 80.25% ^{238}U and no ^{239}Pu is built in. Using this fuel composition, delayed neutrons fractions was calculated for a fresh fuel element. These delayed neutron fractions were compared to the irradiated (old) fuel to estimate burn up. The results are in reasonably good agreement with the experimental data. Based on the calculations fuel element F1 had a total of 2.99 g of Pu built up while F11 was 0.5 and F2 was 0.36 g. These values are within the range of values reported to NRC by the MSTR staff.

MSTR reporting scheme of attributing all the burnup to one fuel element is inherently flawed and must be corrected but this correction will not help much as the previous data is inaccurate and has limited use for calculating burnup on fuel to fuel element basis. The new scheme proposed here can provide reasonable approximation of burnup for each fuel element at MSTR and other MTR reactors.

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Appendix A. Supplementary data

Supplementary data to this article can be found online at https://doi.org/10.1016/j.net.2019.01.005.

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