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Technical Note

A simple method for estimating the major nuclide fractional fission rates within light water and advanced gas cooled reactors

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ABSTRACT

The standard method for calculating anti-neutrino emissions from a reactor involves knowing the fractional fission rates for the most important fissioning nuclides in the reactor. To calculate these rates requires detailed reactor physics calculations based upon the reactor design, fuel design, burnup dependent fuel composition, location of specific fuel assemblies in the core and detailed operational data from the reactor. This has only been published for a few reactors during specific time periods, whereas to be of practical use for anti-neutrino reactor monitoring it is necessary to be able to predict these on the publicly available information from any reactor, especially if using these data to subtract the anti-neutrino signal from other reactors to identify an undeclared reactor and monitor its operation. This paper proposes a method to estimate the fission fractions for a specific reactor based upon publicly available information and provides a database based upon a series of spent fuel inventory calculations using the FISPIN10 code and its associated data libraries.

1. Introduction

Since the first detection of an anti-neutrino from short-lived fission products produced within a nuclear fission reactor in the 1950's [1], the possibility of monitoring a reactor remotely using the characteristic anti-neutrino emissions of a reactor and an inverse beta decay based detector has existed. As detector technology has advanced this has become a potentially practical technology. In the 1990's it was shown that the reactor power and plutonium production could be determined remotely from a power reactor [2]. Since then there has been several experiments to demonstrate and refine monitoring of reactors remotely by this method, see for example the review by Bowden [3].

To implement the technology successfully for out-of-core independent monitoring of a reactor, or for safeguards purposes, it is necessary to understand three areas; (i) the anti-neutrino emission from the reactor being monitored with its relationship to reactor operation, (ii) potential other sources of anti-neutrino background that can interfere with the measurements and (iii) the detector response to anti-neutrinos.

One tool developed to help understand the background anti-neutrino spectra seen at a specific location is the web-based tool https://geoneutrinos.org/reactors/ that allows a detector to be simulated anywhere on the surface of the earth and it then calculates the anti-neutrino rate and its expected spectra given a knowledge of natural geological radioactive deposits and known reactor sources [4]. The tool is based upon the methods used to create global reactor anti-neutrino maps [5,6] supplemented by knowledge of potassium, uranium and thorium deposits [7,8]. The tool uses a standard method to calculate reactor anti-neutrino emission and its spectra that is based upon the widely used postulate that when a reactor is running at constant power the anti-neutrino emission is dependent on the aggregate anti-neutrino spectra from the fission of the most important four fission nuclides in power reactors (^{235}U, ^{239}Pu, ^{238}U and ^{241}Pu) [9,10] which can then be combined using a set of fractional fission rates (FFR) to calculate the emission from individual reactors. The FFR will depend on a reactor design, fuel loading pattern and shuffling, fuel enrichment, fuel burnup and, to a lesser extent, power. However, due to lack of information on individual reactors a review in Ref. [5] was used to determine typical values in light water reactors with a single set being implemented (^{235}U: 0.56,

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^{238}U: 0.08, ^{239}Pu: 0.30, ^{241}Pu: 0.06). Similarly gas cooled reactors are assumed to have a fission fraction of ^{235}U: 0.7248, ^{238}U: 0.0423, ^{239}Pu: 0.2127, ^{241}Pu: 0.0202. The current default values can be seen on the website and updated by the user for calculations within the tool.

In practice, the anti-neutrino emission from a reactor depends on the 3D distribution of fission rates throughout the core. These parameters themselves depend on the fuel composition and neutron flux within the core. This requires a detailed burnup analysis of the reactor including knowledge of the fuel placed within the core with its design, including initial composition, and operations parameters such as boron poison concentration in the water (PWR), void fractions (BWR) and/or control rod insertion into the core, as well as temperatures of the fuel and coolant, and the reactor thermal power. In practice, this is information that is only known to those modelling the reactor to support operation and seldomly distributed to others. It is thus necessary to develop a method to approximate the overall reactor FFR. This paper proposes and describes an initial implementation of such a method.

2. Calculation of the fission fractions using the UK spent fuel inventory code FISPIN

The UK spent fuel inventory code FISPIN10 [11] is distributed with a series of reactor specific libraries based upon reactor physics modelling of various fuel designs and initial enrichments simulating the burnup of uranium dioxide fuel assemblies in AGR, BWR and PWR reactors. These libraries are supplied at different initial ^{235}U enrichments. The reactor physics calculations for these being based upon fuel and reactor design information for the Heysham, Fukushima Daiichi unit 5 and Unterweser reactors respectively. It is recognised that different fuel designs within a reactor and the evolving designs of stations will introduce differences in the results, but this information is seldomly publicly available and for this initial development of an approximate method these representatives of the three reactor classes will be used. The reactor physics models include burnup and self-shielding effects including the depression of the neutron flux due to absorption by fuel components at different neutron energies and the spatial variation of the neutron flux in the moderator and the fuel. The FISPIN libraries contain fuel region averaged cross-sections that maintain the reaction rates in the reactor physics model and follows how these change with fuel burnup.

The FISPIN10 code uses the FFR from fissile and fertile species to calculate the fission product production based upon the correlated neutron flux spectra, instantaneous fuel composition of these nuclides, burnup dependent fission cross-sections, energy per fission and point power. The nine nuclides considered in most FISPIN10 calculations are ^{232}Th, $^{233, 235, 236, 238}$U, and $^{238, 239, 240, 241}$Pu. It is possible for FISPIN10 to report the FFR for these nuclides at the start and end of a burnup step, or within burnup sub-steps.

2.1. FISPIN calculations

To model the FFR with burnup a series of AGR, PWR and BWR calculations were made from zero to 51 GWd per metric tonne of initial uranium (GWd/t) in steps of 1 GWd/t. The AGR assemblies were modelled at initial enrichments of 1, 2, 3 and 4 wt percent (Wt %) at typical powers of 10, 15 and 20 MW/t. The BWR assemblies were modelled at 1, 2, 3 and 4 wt% at typical powers of 10, 20 and 30 MW/t. The PWR assemblies were modelled at 2, 3, 4 and 5 wt% at 20, 35 and 50 MW/t. For the higher enrichments of BWR fuels (3 and 4 wt%) and PWR fuels (4 and 5 wt%) where higher burnups are potentially achievable the maximum burnup was increased to 61 GWd/t. This allowed the generation of a database of FFR from which any AGR, BWR and PWR assembly could have its burnup dependent FFR values estimated. This database is available for download from the Mendeley data archive [12].

An example of the calculated data for the variation of ^{235}U and ^{239}Pu FFR with burnup in an AGR assembly generating 10 MW/t are shown in Fig. 1. This shows the faster growth of ^{239}Pu with lower ^{235}U enrichments to achieve the same fission power with the changing fuel composition, neutron spectra and self-shielding factors.

Fig. 2 shows the FFR for the significant components for an AGR assembly with 1 wt% enrichment irradiated at 10 MW/t. This is a very low enrichment and would not normally achieve burnup values much above 15 GWd/t under normal operating conditions as it does not contribute positively to the neutron economy of the reactor beyond this but is shown up to 50 GWd/t to show the evolution of the major components of the FFR. The FFR is dominated initially by ^{235}U at around 0.95, which then tends towards zero as this nuclide is burnt out. The ^{239}Pu value rises from zero, as this nuclide is produced, tending towards an equilibrium value of around 0.65. The ^{238}U represents an almost constant FFR of 0.05 with ^{241}Pu growing from zero to around 0.25. The other five components, ^{232}Th, ^{233}U, ^{236}U, ^{238}Pu and ^{240}Pu are not shown, and sum to less than 0.004 at the highest burnup.

2.2. Parameterisation of the FFR

Given the size of the database it is useful to develop a parameterisation of the FFR for efficiency. It was found that the principle fractions could be parameterised using the following form:

$$^{238}\mathrm{f} = c + bI + aI^2$$

$$^{239}\mathrm{f} = (1 - \exp(-dI))^e \times f$$

$$^{241}\mathrm{f} = (1 - \exp(-gI))^h \times i$$

$$^{235}\mathrm{f} = 1 - {}^{239}\mathrm{f} - {}^{241}\mathrm{f} - {}^{238}\mathrm{f}$$

Where the FFR values of ^{235}U, ^{238}U, ^{239}Pu and ^{241}Pu are given as ^{235}f, ^{238}f, ^{239}f and ^{241}f and I is the burnup in GWd/t. The parameters *a, b, c, d, e, f, g, h* and *i* are least-square fitted parameters for each modelled case. It is noted that for the same enrichments no changes in the parameters were required over the range of the modelled powers (MW/t) to estimate the trends of the nuclides to the same goodness of fit (typically ± 0.02 in FFR) although there are power related trends resulting from the different times at different powers to reach the same burnup and thus a different amount of the ^{241}Pu decaying. As this ^{241}Pu effect is small, below 50 GWd/t, it was decided for simplicity to ignore this in this work. If the ^{241}Pu differences are shown to be important for a practical application either the full dataset will need to be interpolated or the parameterisation extended to include a rating related parameter to improve the agreement.

The parameters for AGR, BWR and PWR fuels are shown in Tables 1–3. Figs. 3–5 show the agreement between the FFR calculated and this parameterisation. The major differences in the AGR and PWR comparisons with the FISPIN results are a trend away at higher burnups due to the ^{239}Pu and ^{241}Pu parameterisation effects which introduces variations of up to 0.02 in FFR. The parameterisation relies upon a constraint that the four components sum to unity with the ^{235}U being calculated as the difference between unity and the sum of the three explicitly parameterised components. This approach forces the ^{235}U component to reflect

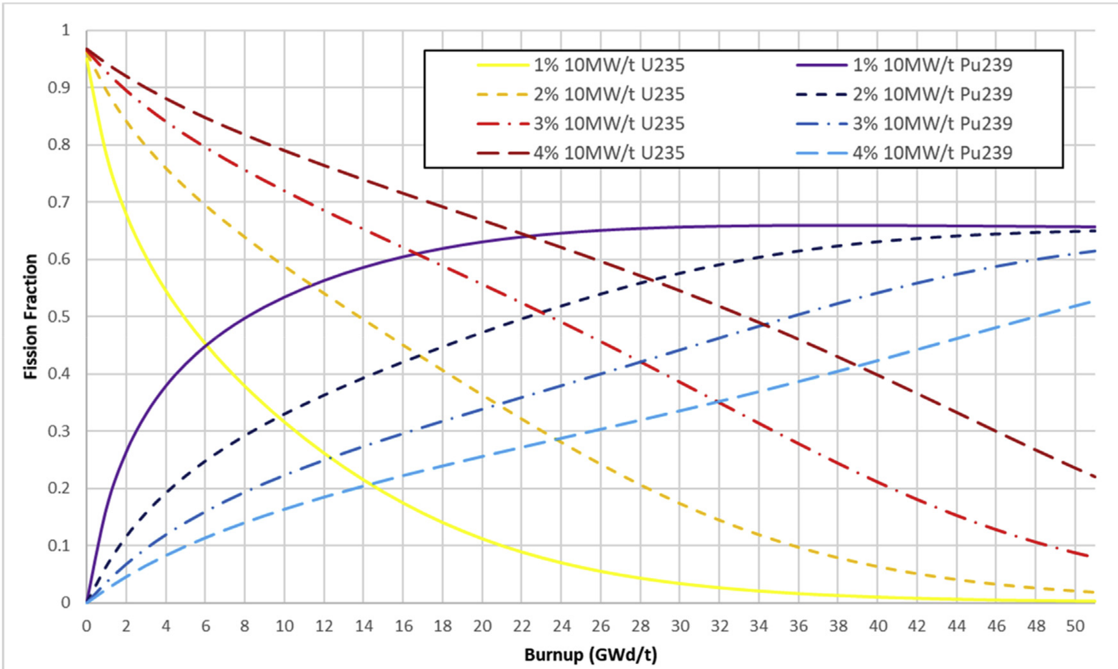

Fig. 1. Fractional fission rates for ^{235}U and ^{239}Pu for AGR assemblies with 1, 2, 3 and 4 wt% enrichments irradiated to generate a power of 10 MW/t.

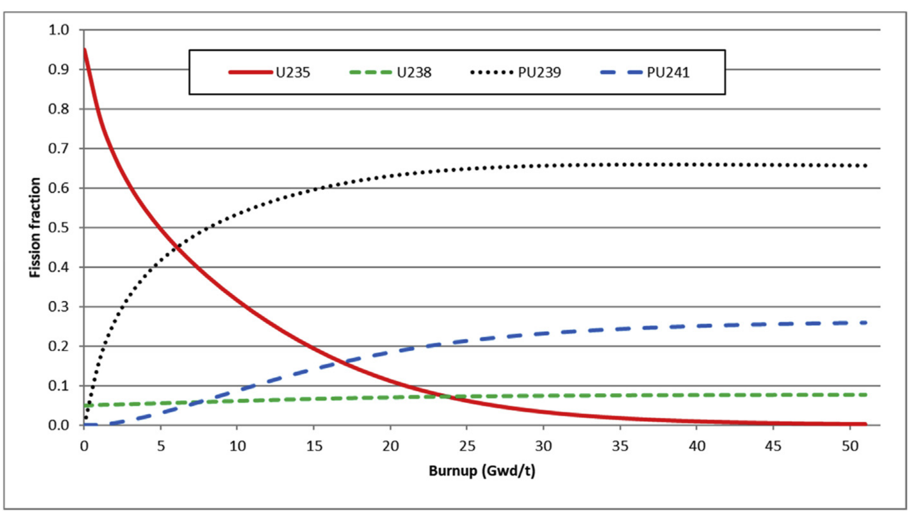

Fig. 2. Variation of all fissionable nuclides with fractional fission rates greater than 0.001 for an AGR assembly of 1 wt% enrichment irradiated at 10 MW/t.

Table 1
AGR parameters for the model at different enrichments.

^{235}U enrichment (Wt%)		^{238}U parameters		^{239}Pu parameters		^{241}Pu parameters
1%	*A*	−1.61598E-05	*d*	1.35639E-01	*g*	1.00506E-01
	B	1.29477E-03	*e*	6.55210E-01	*h*	2.34822E+00
	C	5.01111E-02	*f*	6.58000E-01	*i*	2.61481E-01
2%	*A*	−7.09194E-06	*d*	4.35337E-02	*g*	5.59880E-02
	B	1.14039E-03	*e*	7.44574E-01	*h*	2.57698E+00
	C	3.80158E-02	*f*	7.20137E-01	*i*	3.01026E-01
3%	*A*	4.25275E-06	*d*	1.33134E-02	*g*	3.41233E-02
	B	5.59111E-04	*e*	7.51525E-01	*h*	2.59617E+00
	C	3.51560E-02	*f*	1.03805E+00	*i*	3.73220E-01
4%	*A*	6.84297E-06	*d*	7.04777E-03	*g*	2.63501E-02
	B	2.64411E-04	*e*	7.75717E-01	*h*	2.63440E+00
	C	3.36829E-02	*f*	1.26159E+00	*i*	3.68425E-01

Table 2
BWR parameters for the model at different enrichments.

^{235}U enrichment (Wt%)		^{238}U parameters		^{239}Pu parameters		^{241}Pu parameters
1%	*a*	−3.86986E-06	*d*	5.79969E-02	*g*	6.07264E-02
	b	1.02101E-03	*e*	8.09502E-01	*h*	2.68529E+00
	c	6.13033E-02	*f*	6.64286E-01	*i*	2.54979E-01
2%	*a*	−1.79674E-06	*d*	2.74557E-02	*g*	4.03278E-02
	b	9.34963E-04	*e*	8.14913E-01	*h*	2.69003E+00
	c	5.21111E-02	*f*	7.40594E-01	*i*	2.74202E-01
3%	*a*	1.55961E-06	*d*	1.84379E-02	*g*	3.21997E-02
	b	6.02659E-04	*e*	8.35425E-01	*h*	2.72713E+00
	c	4.94186E-02	*f*	7.56756E-01	*i*	2.60898E-01
4%	*a*	1.01273E-05	*d*	9.55603E-03	*g*	2.37905E-02
	b	7.92128E-05	*e*	8.59253E-01	*h*	2.79432E+00
	c	4.32694E-02	*f*	9.55043E-01	*i*	3.01464E-01

Table 3
PWR parameters for the model at different enrichments.

^{235}U enrichment (Wt%)		^{238}U parameters		^{239}Pu parameters		^{241}Pu parameters
2%	*a*	−1.03853E-05	*d*	7.78433E-02	*g*	7.40333E-02
	b	1.26634E-03	*e*	8.23008E-01	*h*	2.81525E+00
	c	6.61256E-02	*f*	6.16124E-01	*i*	2.30278E-01
3%	*a*	−3.98861E-06	*d*	4.89851E-02	*g*	5.27438E-02
	b	9.49603E-04	*e*	8.62627E-01	*h*	2.80454E+00
	c	5.93779E-02	*f*	5.97942E-01	*i*	2.24597E-01
4%	*a*	6.23551E-07	*d*	3.74155E-02	*g*	4.30585E-02
	b	6.09234E-04	*e*	8.89526E-01	*h*	2.84277E+00
	c	5.66330E-02	*f*	5.66078E-01	*i*	2.09560E-01
5%	*a*	4.99332E-06	*d*	3.19341E-02	*g*	3.86218E-02
	b	2.77454E-04	*e*	9.12152E-01	*h*	2.93710E+00
	c	5.57676E-02	*f*	5.24947E-01	*i*	1.90159E-01

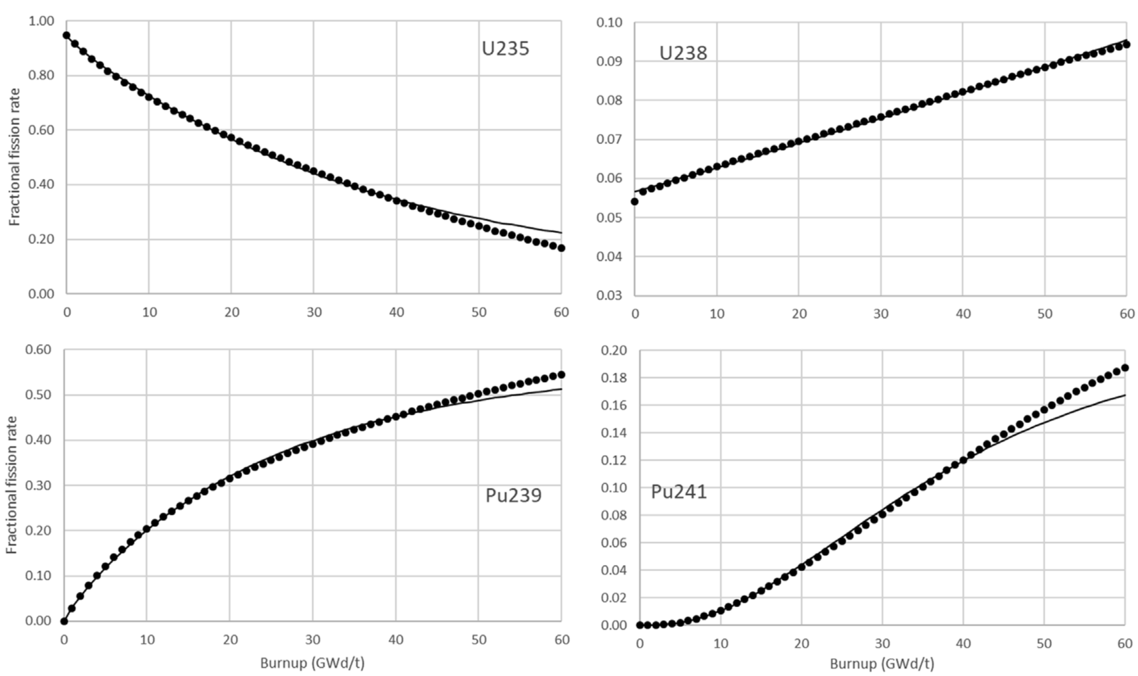

Fig. 3. Principle fraction fission rates of ^{235}U, ^{238}U, ^{239}Pu and ^{241}Pu for a PWR 4% enriched UO_2 fuel assembly irradiated at a rating of 20 MW/t. Markers showing FISPIN values and lines the results of the parameterisation.

the combined differences. However, it is expected that for anti-neutrino applications that the anti-neutrinos per fission from these nuclides will be similar so the overall effect would be around 2% in anti-neutrino total emission. However the shape of the anti-neutrino energy spectra may weight the sensitivity of a measurement. If an experiment is dominated by the ^{241}Pu components it could show differences using this parameterisation of up to 20% and interpolation of the database would be needed. The BWR results show similar trends, but a significant trend away from the model occurs below 8 GWd/t where gadolinium burnable poison pins are present in the assembly design. It is noted that the effect of such burnable poison pins becomes negligible above burnups of 8

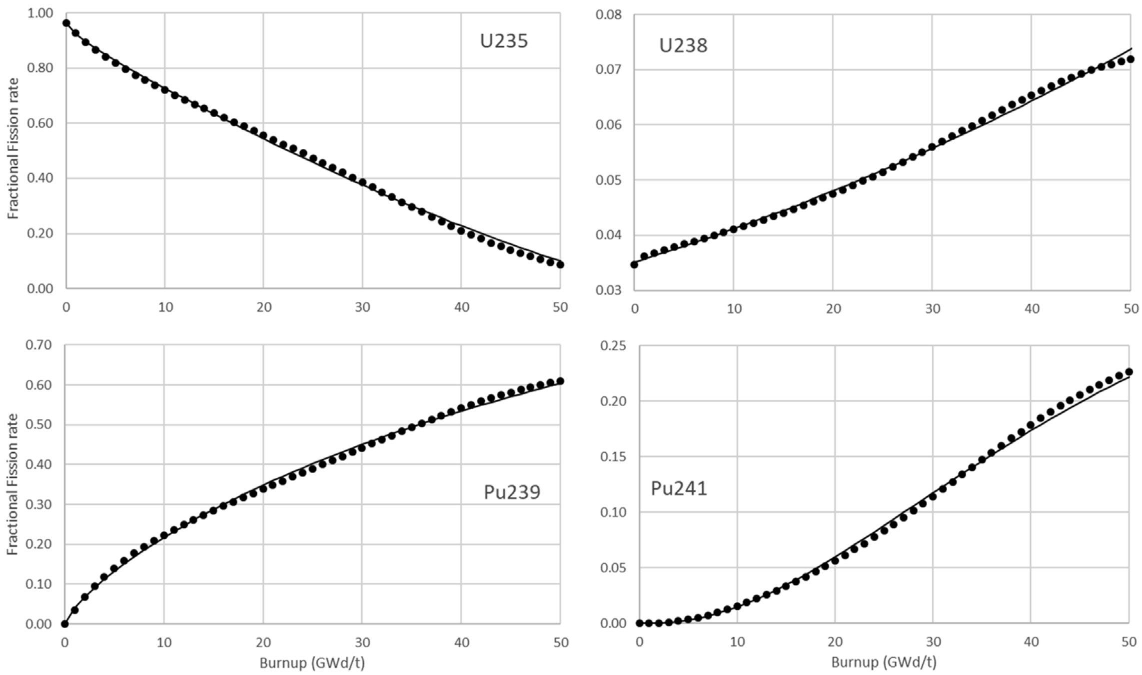

Fig. 4. Principle fraction fission rates of ^{235}U, ^{238}U, ^{239}Pu and ^{241}Pu for a AGR 3% enriched UO_2 fuel assembly irradiated at a rating of 10 MW/t. Markers showing FISPIN values and lines the results of the parameterisation.

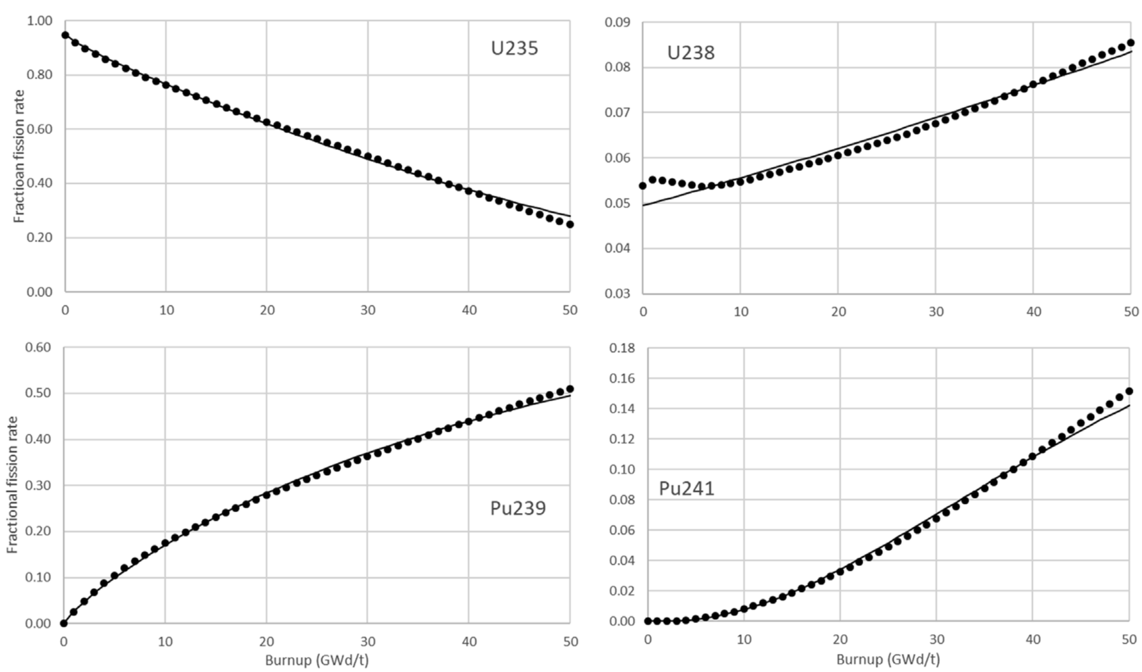

Fig. 5. Principle fraction fission rates of ^{235}U, ^{238}U, ^{239}Pu and ^{241}Pu for a BWR 3% enriched UO_2 fuel assembly irradiated at a rating of 10 MW/t. Markers showing FISPIN values and lines the results of the parameterisation.

GWd/t.

3. Methodology for calculation of the core averaged fission fractions using a minimum reactor operational information

As described above a determination of the FFR for a whole reactor core would require a full neutron transport, fuel burnup and thermal hydraulic solution within the reactor based upon the reactor and fuel designs and the setup and operation of the reactor to determine the fission rates of each nuclide in each region at a specific time which could then be summed to give the total rates in the core and thus the FFR at that time. This is a very time consuming calculation relying upon a large amount of information from the reactor operators, which are not usually available. This work proposes a method that uses only publicly available information from the Nuclear Engineering International magazine's World Nuclear Industry Handbook. It relies upon some very significant approximations. These include: that.

- reactors of the same type can be approximated by a single "typical" model
- the fuel assemblies can be modelled as independent units that are little affected by the surrounding fuel assemblies (i.e. the assemblies can be modelled as a single assembly with reflective boundary conditions)
- the fuel can be modelled as a 2D slice through the assembly with average temperatures, densities and powers (i.e. using averages for these parameters do not affect the overall whole assembly results)
- neutron absorption controlling the reactor; including control rods, boron in the moderator and burnable absorbers do not significantly alter the overall fission rates
- assemblies placed within the core at a certain time can be considered as a group modelled using the reactor average power during their irradiation

If we make these assumptions then the default libraries in FISPIN, or similar systems such as ORIGEN ARP [12], can be used to estimate the overall FFR in a reactor without attempting a more complex solution requiring more information. It is expected that AGR and PWR, due to only small changes in the moderator density and minimal use of burnable poisons, would be better modelled using these approximations than BWR where there are significant changes to moderator density in the fuel channel and extensive use of burnable poisons as well as the use of control blades to alter the power distribution in the core axially during burnup. It is recognised that some nuclides which are not produced, or destroyed, proportionally with burnup, may not be well modelled by this type of procedure.

Using these assumptions it is possible to consider identical fuel assemblies (specifically design and initial enrichment) placed within the core as independent batches contributing towards the total FFR in proportional to their average burnup by their initial uranium mass, or if all having the same initial uranium mass, their number. The core FFR then becomes a weighted average of the FFR for each fuel batch as it is irradiated.

The above FISPIN modelling shows the effect of enrichment for three commercial power reactor types across a range of fuel enrichments and powers. However, a methodology is required to use this to estimate the nuclide FFR for any arbitrary reactor of these types using the above assumptions.

Reactors generate power from fission in the nuclear fuel assemblies placed within them, this relies on at least one neutron per fission giving rise to another fission to maintain a chain reaction. As the fissile content of fuel is reduced and neutron absorbing products increase the probability of a neutron producing a subsequent fission is reduced. To continue to generate power the fuel which can no longer contribute to maintaining the chain reaction must be replaced. In the reactor types considered here this is done by shutting down the reactor and replacing a fraction of the fuel that has the lowest ability of maintain the chain reaction with fresh fuel. The reactor is then restarted and run until the reactor can, again, no longer maintain the chain reaction. The ability of a fuel assembly to maintain a chain reaction is reduced during burnup and thus the highest burnup fuel assemblies are usually replaced. If we consider the approximation that fuel placed in a reactor core of enrichment, E_F, is irradiated at a rating of R_F [MW/t] until it reaches a discharge burnup, B_f [MWd/t], in a series of n on-power cycles with shutdown periods between, so that after n cycles it would no longer contribute positively to the neutron economy of the core and further that a certain fraction of the fuel is replaced each cycle, F_R, to maintain operation. It then follows that the fuel is irradiated from zero to B_f in these n cycles. So that after the first cycle the fuel has a burnup of B_f/n, the second cycle $2*B_f/n$ and in the third cycle $3*B_f/n$, …, up to the nth cycle. The mean R_F [MW/t] can be calculated by dividing the thermal power of the reactor P_T [MW] by the initial uranium mass in the core M_C [t], although if using the above parameterisation which ignores power this is unnecessary provided information is available on the number of cycles during which the fuel is irradiated. Using the above assumptions, the core at any time will contain n batches of fuel assemblies each batch having been inserted during different shutdowns; the newest having seen zero to one, the second newest one to two, …, and the oldest n-1 to n on-power cycles of irradiation. The average burnup of the core can then be calculated from the n batches of fuel present in the core. In this approximation the mean burnup will begin an on-power cycle at $(1 + 2 + \ldots + n\text{-}1)\ B_f/n$ [GWd/t] and rise uniformly to $(1 + 2 + \ldots + n)\ B_f/n$ [GWd/t].

It should be noted that because the fractional fission rates are not proportional to burnup using the core average burnup value to estimate the core overall FFR values would not accurately represent the core. In this work it is proposed to interpolate the FFR data [13] for each batch of fuel placed within the reactor at each cycle during its irradiation and by averaging the values for each batch determine an average value during an on-power period. To do this it is necessary to interpolate the database or the parameterisation for the governing parameters; enrichment, power (if not using the simplified parameterisation) and burnup, which can be accomplished using trilinear interpolation methods. The easiest method to achieve this is to first interpolate the data to the known enrichment resulting in a two parameter database dependent upon power and burnup, then interpolate the data to the average power resulting in a database dependent on burnup. The FFR values can then be interpolated to the required burnup. It is noted that the fuel assemblies in some reactors are deliberately shuffled into different regions of the core to reduce power peaking which can introduce considerable different assembly powers during each cycle. If the burnup distribution was available for the whole core, including every fuel assembly and axially along each of these it would be possible to better represent the core, although this information is seldomly available except to the reactor operators.

A reactor running at the same power using the same fuel design and enrichments without any need to change any operational procedures to improve safety or economic efficiency, would approach an optimal refuelling strategy similar to the above approximation. However, it is expected that reactor operators wanting to increase power, reduce the percentage of shutdown time or extend the reactor life would need to vary the number and type of assemblies loaded. The study of these effects are beyond the scope of this initial work, but would need to be studied to validate the accuracy of this assumption.

4. An example using available data from Daya Bay to calculate whole core FFR

From the literature the information on the Daya Bay 1 reactor

Table 4
Daya Bay 1 reactor information [14,15].

Quantity	value	Unit	Reference
E_F	4.0	Wt% ^{235}U/U	An, 2016
R_F	$B_f/M_c = 40.124$	MW/t	–
B_f	43000	MWd/t	PRIS, 2019
N	3	–	An, 2016
F_R	1/3	–	An, 2016
P_T	2905 MW	MW	PRIS, 2019
M_C	72.4	t	PRIS, 2019

Table 5
FISPIN database fission fraction values for PWR 4 wt% at 0 GWd/t irradiated at 20, 35 and 50 MW/t.

Nuclide	20 MW/t	35 MW/t	50 MW/t	Interpolate to 40.124 MW/t
Th232	0.000E+00	0.000E+00	0.000E+00	0.000E+00
U233	0.000E+00	0.000E+00	0.000E+00	0.000E+00
U235	9.459E-01	9.459E-01	9.422E-01	9.440E-01
U236	2.587E-05	2.587E-05	2.764E-05	2.676E-05
U238	5.412E-02	5.412E-02	5.774E-02	5.594E-02
Pu238	0.000E+00	0.000E+00	0.000E+00	0.000E+00
Pu239	0.000E+00	0.000E+00	0.000E+00	0.000E+00
Pu240	0.000E+00	0.000E+00	0.000E+00	0.000E+00
Pu241	0.000E+00	0.000E+00	0.000E+00	0.000E+00

Table 6
FISPIN database fission fraction values for PWR 4 wt% at 43 GWd/t irradiated at 20, 35 and 50 MW/t.

Nuclide	20 MW/t	35 MW/t	50 MW/t	Interpolatedto 40.124 MW/t
Th232	1.173E-11	6.702E-12	4.334E-12	6.325E-12
U233	1.478E-07	1.409E-07	1.024E-07	1.226E-07
U235	3.126E-01	3.121E-01	1.935E-01	2.524E-01
U236	1.266E-03	1.264E-03	1.111E-03	1.187E-03
U238	8.417E-02	8.402E-02	9.278E-02	8.846E-02
Pu238	4.804E-04	4.237E-04	4.951E-04	4.689E-04
Pu239	4.683E-01	4.666E-01	5.371E-01	5.024E-01
Pu240	1.404E-03	1.402E-03	1.670E-03	1.537E-03
Pu241	1.319E-01	1.343E-01	1.733E-01	1.536E-01

was obtained and is give in Table 4. The World Nuclear Industry Handbook has not be published for several years, and thus to ensure current information the data was obtained from the IAEA PRIS database [14] supplemented with a recent publication on the Daya Bay 1 reactor [15]. As the anti-neutrino measurements were carried out over several cycles it is possible to consider only the mid-point of each cycle to give a mean set of FFR values. Considering the irradiation of a batch of fuel in the reactor, during its first cycle it is irradiated from zero to a 14333 MWd/t burnup with a mid-point of 7167 MWd/t. In the second cycle from 14333 to 28667 with a mid-point of 21500 MWd/t. In the final cycle from 28667 to 43000 with a mid-point of 35833 MWd/t. Taking the database values for 4 wt% PWR fuel for burnups we can interpolate to the mean core power of 40.125 MW/t. This is shown in Tables 5 and 6 for the beginning (zero MWd/t) and end of the irradiation of a specific fuel reload (43000 MWd/t).

If we apply this same method for the database values nearest to the mid-points of the irradiation cycles we get the values in Table 7. If we now interpolate the values in Table 7 to the mid-point burnups of each cycle we obtain the values in Table 8. As in any irradiation cycle a third of the fuel is in its first cycle, a third is in its second cycle and a third is in its third and final cycle, the whole core can be estimated as the mean of these mid-points.

It is noted in this case that the enrichment coincided with a value in the database. In the general case, the above would need to be carried out using a trilinear interpolation method.

This gives the fractional fission rates of ^{235}U: ^{238}U: ^{239}Pu: ^{241}Pu as 0.521 : 0.074; 0.339 : 0.064 respectively (noting that these do not sum to unity due to small components of other fissioning nuclides in Table 8) which compares to the Daya Bay reported values in Ref. [15] of 0.561 : 0.076: 0.307 : 0.056 respectively with uncertainties conservatively estimated as 5%. An uncertainty on the values calculated using the method in this work could potentially be obtained by comparing the accuracy of the major plutonium and uranium species predictions with FISPIN post irradiation spent fuel analyses and the uncertainty on the fission cross-sections. Such future work could help validate the method. It is noted that these FFR value differences are larger that the quoted Daya Bay uncertainties, the differences being 10–20%. The effect of these differences on the analysis of an anti-neutrino experiment would depend on the sensitivity of the experiment to the FFR values.

Table 7
FISPIN database fission fraction values for PWR 4 wt% during irradiation using interpolation to actual power of 40.124 MW/t.

Nuclide	Fractional fission rates as a function of burnup							
	0 MWd/t	7000 MWd/t	8000 MWd/t	21000 MWd/t	22000 MWd/t	35000 MWd/t	36000 MWd/t	43000 MWd/t
Th232	0.000E+00	2.026E-13	2.592E-13	1.616E-12	1.767E-12	4.295E-12	4.532E-12	6.325E-12
U233	0.000E+00	3.831E-08	4.301E-08	9.041E-08	9.310E-08	1.169E-07	1.180E-07	1.226E-07
U235	9.440E-01	7.443E-01	7.228E-01	5.042E-01	4.903E-01	3.323E-01	3.216E-01	2.524E-01
U236	2.676E-05	2.804E-04	3.135E-04	7.027E-04	7.292E-04	1.036E-03	1.057E-03	1.187E-03
U238	5.594E-02	6.367E-02	6.445E-02	7.385E-02	7.453E-02	8.332E-02	8.398E-02	8.846E-02
Pu238	0.000E+00	3.892E-06	5.482E-06	6.917E-05	7.843E-05	2.744E-04	2.956E-04	4.689E-04
Pu239	0.000E+00	1.847E-01	2.029E-01	3.616E-01	3.702E-01	4.604E-01	4.662E-01	5.024E-01
Pu240	0.000E+00	1.278E-04	1.582E-04	6.471E-04	6.882E-04	1.227E-03	1.267E-03	1.537E-03
Pu241	0.000E+00	6.914E-03	9.356E-03	5.895E-02	6.344E-02	1.215E-01	1.256E-01	1.536E-01

Table 8
FISPIN database fission fraction values for PWR 4 wt% during irradiation at 40.124 MW/t interpolated to the cycle mid-point burnup values.

Nuclide	Fractional fission rates as a function of burnup			Average of the values
	7166.667 MWd/t	21500 MWd/t	35833.333 MWd/t	
Th232	2.120E-13	1.692E-12	4.493E-12	2.132E-12
U233	3.909E-08	9.176E-08	1.178E-07	8.289E-08
U235	7.407E-01	4.973E-01	3.234E-01	5.205E-01
U236	2.859E-04	7.160E-04	1.054E-03	6.851E-04
U238	6.380E-02	7.419E-02	8.387E-02	7.395E-02
Pu238	4.157E-06	7.380E-05	2.921E-04	1.233E-04
Pu239	1.877E-01	3.659E-01	4.652E-01	3.396E-01
Pu240	1.329E-04	6.677E-04	1.260E-03	6.870E-04
Pu241	7.321E-03	6.120E-02	1.249E-01	6.448E-02

5. Conclusion

The Daya Bay reactor operators reported fractional fission rates from a whole core calculation based upon the reactor design, actual fuel loading patterns, fuel designs and actual operational information using their own codes and underpinning nuclear data [15] giving a similar ordering of FFR components to the much simpler model proposed in this work but with up to 10–20% differences, which are much larger that the Daya Bay reported uncertainties implying that improved methods are needed to estimate the FFR values for any given reactor and this may require more information from the reactor to be incorporated. The FISPIN10 results presented for the Daya Bay conditions shows that the ^{235}U FFR decreases during irradiation from 0.94 to 0.25 and the ^{239}Pu values increase from zero to 0.50. Thus the crude agreement with the Daya Bay reported results suggests that this simplified model may be useful in supplying FFR values trends for reactors to those developing the technology to use anti-neutrino emissions to monitor reactors, but improvements are necessary before it can be quantitatively useful. Such trends in the fractional fission rates could assist in investigating the effect of different operating conditions such as changing the enrichment of fuel, the average power, final irradiation or number of cycles that the fuel is resident.

Declaration of competing interest

The authors declare that they have no known competing financial interests or personal relationships that could have appeared to influence the work reported in this paper.

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Appendix A. Supplementary data

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