The use of interval calculation technique for fuel characteristic uncertainty estimations into a fuel cycle

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Estimation of isotope concentrations in a fuel during its operation into reactor and into spent fuel storage, and also an estimation of nuclear and radiating safety characteristics in these systems are carried out by use of various methods and codes. But practically all these codes are either deterministic or Monte-Carlo ones, so it is impossible to define the correct uncertainties attributed to characteristics derived.

That is why for deriving of correct estimations of fuel characteristics uncertainties (in this case for a nuclide concentration) during fuel life cycle the new technique, based on application of a special interval calculations technique has been created [1].

Basically, the problem of dependency of fuel cycle characteristic uncertainties from source group constants and decay parameters uncertainties can be solved (to some extent) as well by use of sensitivity analysis. However such procedure is rather labor consuming and does not give guaranteed estimations for received parameters since it works, strictly speaking, only for small deviations cause it is initially based on linearisation of the mathematical problems.

Suggested and realized technique of fuel cycle characteristics uncertainties estimation is based on so-called interval analysis (or interval calculations). The basic advantage of this technique is the opportunity of deriving correct estimations**.** In a professional terms this decision consist on introduction of a new special type of data such as Interval data in a codes and definition for them all arithmetic operations [2-3].

Interval type data are in a real practice operation and use now. There are many realizations of interval arithmetic implemented by different ways.

Authors realized a technique of the ***Cauchy*** problem decision for system of the linear equations (isotope kinetics) with use of interval arithmetic for the fuel burning up problem. Thus there is an opportunity to research a neutron flux, fission and capture cross-section uncertainties impact (and also of nuclide yield and decay constants) on nuclide concentration uncertainties and, accordingly, on change of a fuel cycle characteristics (such K_{eff}, breeding ratio etc).

As an example the standard WWER-1000 reactor three-zoned cell has been calculated. Cell parameters were as following: fuel radius is 0.39 sm, external clad radius is 0.455 sm, moderator radius is 0.66942 sm with initial fuel concentrations equal to $7.642*10^{20}$ nucl. / sm3 for ^{235}U and $2.24228*10^{22}$ nucl. / sm3 for ^{238}U. One-group cross-sections were calculated by the Monte-carlo code with corresponding. In our simulation problem an uncertainties for one-group constants have been accepted according to author's uncertainty estimations [4]. Calculations were carried out for neutron flux of $4*10^{14}$ n/sm2*sec and fuel campaign duration equal to one and three years.

We also carried out research of sensitivity of concentration uncertainties to uncertainties of neutron flux and uncertainties of the initial cross-section data. Calculations were carried out for neutron flux uncertainties of 1 % and 5 % and for uncertainties in the initial cross-section data of 1 % for ^{235}U and ^{238}U. Results are presented for one and three years of a reactor operation.

Thus, the presented technique allows us to derive correct guaranteed estimations of uncertainties. In some cases they can be overestimated, but not always. As an example we carried out comparison of concentration values for ^{242}Am and ^{239}U at operation time T=1 year, neutron flux

(4.00+/- 0.04)* 10^{14} n/sm2*sec and for precisely set flux value of $4.04*10^{14}$ n/sm2*sec. As a result for ^{242}Am the top bound appeared to be overestimated, and for ^{239}U - is not overestimated.

Application of this technique is especially interesting at an estimation of perspective reactors characteristics of in general and systems with complex isotope structure in particular. Namely for an assessment of critical molten salt reactor properties we carried out an estimation of this system characteristics and their uncertainties.

We carried out an estimation of uranium, plutonium and actinides concentration, and also their uncertainties, into the core of molten salt rector with salts of these elements. As aprioristic uncertainty estimation of nuclide cross-sections we use the estimations similar to resulted in [4]. Results the reactor operation time equal to T=1 year are presented at the Table 1.

Table 1. Variation of nuclei concentration and their uncertainties into the core of molten salt reactor for neutron flux 10^{13} and 10^{15} n/sm2*sec

Izotope	Start Concentration	Flux $=10^{13}$ n/sm2*sec			Flux $=10^{15}$ n/sm2*sec		
		Fin Concentration	Uncertainty	Uncertainty %	Fin Concentration	Uncertainty	Uncertainty %
U235	5,992E+16	6,065E+16	9,056E+12	0,01	8,345E+16	9,970E+14	1,19
U236	4,049E+16	4,123E+16	4,106E+12	0,01	4,232E+16	4,738E+14	1,12
U238	1,925E+15	1,930E+15	1,897E+11	0,01	1,638E+15	1,623E+13	0,99
Np237	3,596E+18	3,602E+18	1,699E+15	0,05	1,872E+18	9,022E+16	4,82
Pu238	2,363E+19	3,667E+19	7,793E+15	0,02	2,869E+19	6,625E+17	2,31
Pu239	7,518E+18	7,608E+18	3,754E+15	0,05	8,466E+18	3,137E+17	3,71
Pu240	5,711E+18	8,301E+18	6,764E+15	0,08	6,171E+18	4,875E+17	7,90
Pu241	2,896E+18	2,787E+18	5,569E+15	0,20	3,580E+18	4,559E+17	12,73
Pu242	6,101E+18	6,139E+18	4,528E+15	0,07	6,185E+18	4,599E+17	7,44
Am241	1,743E+19	1,737E+19	7,192E+15	0,04	6,523E+18	2,776E+17	4,26
Am242m	4,994E+17	4,860E+17	6,685E+14	0,14	1,349E+17	1,001E+16	7,42
Am242	1,395E+17	4,121E+14	1,644E+13	3,99	1,513E+16	1,281E+15	8,47
Cm242	1,674E+19	3,635E+18	5,251E+15	0,14	6,171E+18	5,475E+17	8,87
Cm243	2,406E+18	2,330E+18	1,094E+16	0,47	1,462E+18	6,860E+17	46,94
Cm244	6,977E+19	6,728E+19	8,459E+16	0,13	6,158E+19	7,878E+18	12,79

As a result, concentration uncertainties is seriously incriased for higher flux value, these uncertainties are lower for isotopes with well-known cross-sections and they are relatively low for rare isotopes(with low concentrations).

CONCLUSION

So, interval calculation technique is proved to be useful for reactor fuel uncertaities calculations. Nowdays this technique is developing for thermahydrawlic problem and may be economic ones.

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