calculations of discharged fuel composition - the next sum rule (mass-energy conservation law) must be observed independently from chosen nuclear data values and reactor parameters: mass of charged fuel is equal to sum of mass of discharged fuel, produced energy in mass units, loosed neutrons mass and mass of produced He. The actinides mass at storage at any moment of time together with mass of decay products must be equal to initial mass of actinides in the storage. It is shown that the first rule is not fulfilled in CURE calculations using ORIGEN2 code for composition of PWR spent fuel, and the second - in calculations of Am-enriched fuel in storage using KARE code. Source of such discrepancies needs thorough investigation.

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STATUS OF RUSSIAN DOSIMETRY FILE

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Russian Dosimetry File (RDF) containing 35 original dosimetry reaction cross sections evaluated at the Russian Nuclear Data Center(CJD) is described in the present work.

List of the reactions covers a set of dosimetry reactions used in fission reactors and some reactions of important for fusion reactor dosimetry. The data are given in the neutron energy region up to 20 MeV. The evaluated cross sections and their covariences are presented in ENDF-6 format (Files 3, 30, 33).

The reliability of the evaluations from RDF was confirmed by comparison with average cross sections measured in Cf-252 and U-235 fission spectra, fast reactor, T(d,n) and Li(d,n) neutron fields.

EXAMINATION OF NP-237 AND AM-241 NEUTRON DATA ON FAST CRITICAL ASSEMBLIES EXPERIMENTS

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On the series of fast critical assemblies BFS in Russia (BFS-67 and BFS-69) and FCA in Japan Np-237 and Am-241 central fission ratios and reactivity rates were measured. In the fast reactor BN-350 in Kazakhstan ratios of capture cross sections of these nuclides were measured in different places of the core. The totality of these experimental data allows examining used neutron cross sections of mentioned actinides, which are the more major among the minor actinides. It must be pointed out that these data were obtained for different neutron spectra and thus allow examining acceptability of cross sections for wide range of fast reactors. Majority of experimental data was obtained by using the samples of the not negligible small dimensions in the heterogeneous cores. Thus sophisticated evaluation of the raw experimental data was necessary before comparison with calculations. Process of evaluation is briefly considered in the report. Evaluated data with obtained corrections and their uncertainties are given. All these data are compared with results of calculations performed using the modern neutron data evaluations (BROND-2, ENDF/B-6, JENDL-3 and JEF-2) for examined nuclides and the ABBN-90 group cross section set for assemblies' materials. Results of comparison are discussed.