System of Codes and Nuclear Data for Neutronics Calculations of Fast Reactors and Uncertainty Estimation

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Abstract. The paper considers a modern state of Russian neutronics computer codes used in fast reactor applications for definition the core and nuclear cycle parameters. ROSFOND evaluated nuclear data files and ABBN group data set are used as a basis for nuclear data. The main feature of all codes is that they use one and the same, unique constants data base ABBN with the code CONSYST for generation of effective cross-sections. INDECS system of codes and archives is used for uncertainty estimations which is based on usage of perturbation theory and covariance matrices of nuclear constants. TRIUM code based on GRS method is now developed. It is a synthesis of engineering codes and INDECS system.
Key Words: CONSYST/ABBN, INDECS, codes for fast reactor calculation and uncertainty analysis.

1. Introduction

Designing of neutronics characteristics of fast reactor cores and fuel cycle requires to use certified, qualified sets of codes and nuclear constants. The calculation tools should be related to the modern state of scientific knowledge and computational techniques. The used nuclear physics constants should be adequate to the most reliable evaluations of nuclear data.

In connection with rapid development of the computing engineering, and especially personal computers, and all greater introduction in practice of calculation codes on the basis of method of Monte Carlo, the methodical constituent of calculation error falls down substantially. In these terms the constant constituent of error of calculations becomes fully qualificatory. A situation is intensifyed by the fall-off of financing of experimental works, why in this connection the amount of fast critical stands in the world diminishes sharply.

The paper analyzes the state of art in constants supplying in relation to fast reactors with a closed fuel cycle.

2. Modern State of the ABBN Constants System

Almost from the first attempts to create a fast neutron reactors in the practice of settlements began to be introduced electronic computers and the prerequisites for the development of the microscopic theory of reactors based on multigroup approximation. Being technically more simple, but requires a large amount of raw data and large neutron-physics calculations, the implementation of multi-group microscopic theory helped to solve the problem of neutronics calculations of fast and intermediate reactors. In 1962 was developed and in 1964 published the first version of Russia ABBN 26-grouped constants data set [1]. It was differed from others by big enough amount of nuclides used in the reactor industry. The number of energy groups was enough to describe neutron spectra in fast and intermediate energies in reactor cores of different types. The main achievement was an approach of “Bondarenko f-factors” for accounting the resonance self-shielding in cross-section’s structure. It was widely used as in Russia so abroad for reactor and shielding calculations. In 1978 ABBN constants system was renewed and extended from 10.5 MeV to 15 MeV [2]. The ABBN-78 was widely propagated and adopted as a standard in Minatom and used till now.

The next ABBN version appeared in 90th was significant improved and quite differed from the previous ones. In 1995 the ABBN-93 [3] was certificated as a recommended data. Creating the new version of the constants ABBN-93 was preceded by the formation of a library of evaluated neutron data files FOND-2.2, which were selected by the most reliable files of evaluated data from the latest versions of libraries BROND, JENDL-3, ENDF / B-6 and JEF-2. For major reactor materials adopted 299 fine-group approach by dividing the traditional ABBN groups for 6-12 narrower ones. Data tables are presented in a uniform format suitable for computer processing as well as for visual viewing. A more detailed partition group in the thermal region allowed entering the scattering "thermalization" matrices. The system included a much broader set of data, such as data on energy release in various neutron reactions of photons produced in these reactions, data on the characteristics of delayed neutrons, sections of neutron reactions, fission product yields, characteristics of the radioactive decay of nuclei formed in neutron reactions and others. The ABBN-93 constants system has been verified to check the internal consistency of the data; calculations have been validated in a large number of macro experiments with different neutron spectra, peer-reviewed and certified as a recommended reference data. The ABBN-93 is a basis for creation of problem oriented data libraries. As an example, in 2003 was developed an ABBN-BREST data sets for calculation of the fast reactor core with the lead coolant.

In 2005 - 2006 have been started a work for creation the first version of new Russian library of evaluated nuclear data files ROSFOND. It now contains about 700 files of data for different reactor materials. The selection of files for RUSFOND was made based on ENDF/B-VI.8 and VII.0, JEF-2.2 and JEFF-3.1, JENDL-3.3, FOND-2.2 and BROND by comprehensive study of their quality. Currently the version ROSFOND-2010 [4] has been developed and based on this the new version ABBN-RF [5] was constructed. Now the ABBN-RF version of constants passes verification and validation in numerous test calculations [6-8].

3. Treating and Preparing Constants for Neutronics Calculations

For treating the ABBN data a special code system CONSYST/ABBN was developed. Modern CONSYST program [9] allows to receive a full set of macroconstants for all reactor zones or (and) the shielding for the calculation of neutron or (and) the photon fields in a variety of approximations as diffusion and P_1, transport approximation with averaged total cross section in zero or first harmonic of the flux, considering scattering anisotropy up to P_5. Effective macroconstants are calculated accounting resonance self-shielding in 299-group energy approximation. The resulting group constants can be folded into a smaller number of groups with weighing zones integral spectra which can be calculated either within the CONSYST program itself or introduced from the outside, for example, after a preliminary calculation using 1D or 2D-dimensional 299-group program. The calculated by the CONSYST program constants can be given in various formats. The most common are: (a) the format ARAMAKO to which is tied a lot of Russian engineering codes, (b) a well-known format ANISN, which (apart from the program of the same name) is tied many other codes, such as KATRIN, DORT and TORT, and (c) AMPX format especially used in the case of Monte Carlo Russian code MMKK [10]. For Russian codes a PRECONS program was created to prepare constants in the ARAMAKO format.

It should be emphasized that the verification and validation of constants of the modern systems ABBN-93 and ABBN-RF is conducted under the condition of their processing by the CONSYST code.

4. Codes for Neutronics Calculations

Codes which are used in the design calculations mostly solve the Boltzman transport equation in diffusion approximation, they are: TRIGEX, JAR-FR, GEFEST, FACT-BR. TRIGEX and JAR-FR codes are widely used in the design calculations of BN reactors (BN-600, BN-800 and BN-1200, BOR-60, MBIR, CEFR), as well as for the analysis of innovative projects of fast reactors. They designed for three-dimensional calculations of fast reactors in a hexagonal geometry, but also allow to count a square lattices. The diffusion equation is solved by the finite difference method or nodal method. The number of energy groups is 26 or 28, the number of described settlement areas - up to 60,000. The TRIGEX essential feature is the presence of a cell module FFCP which allows using the code for the analysis of experiments on critical assemblies BFS to account their complex heterogeneous structure. The codes implemented solution of homogeneous direct and conjugate problems, as well as with an external source, implemented algorithms for calculating the classical first-order perturbation theory, the generalized perturbation theory to calculate the sensitivity coefficients of linear fractional functionals. The FACT-BR code is the main tool used in the physics calculations of the lead-cooled reactor BREST-300. The GEFEST code designed for performance calculations of reactors BN-600 and BN-800 in BNPP in diffusion approximation in the three-dimensional hexagonal geometry. The code allows calculating the neutron fields and power distribution at all points of the reactor model, taking into account the movement of the control rods. The basic calculations are performed in two group approximation, for neutronics calculations the GEFEST complex provides the use of MMKK code. An important part of the complex is the fuel archive containing passport data FA, their characteristics, reflecting the burning mode during the tenure of the assembly in the reactor. To calculate the effects of reactivity in the complex algorithms implemented perturbation theory. It is possible to solve the kinetic equations in the quasi-static approximation. Required parameters of the equation for the amplitude function (lifetime of prompt neutrons and parameters of delayed neutrons) are defined in the complex.

Codes, which are based on Monte-Carlo method, are developed during many years. Nowadays, they have additional impulse in interest due to fast developing of the computation technique. Among Russian codes, as MMKFK and MCU, American codes KENO and MCNP are now widely used for test calculations. Recently, based on MMKFK, a hybrid complex MMK-C [11] was developed. It now widely used in planning and analyzing of reactor-physics experiments as well as for precise calculations of fast reactors BN-600 and BN-800. The MMK-C code includes two independent computation modules: MMKK - group version (basic version had the name MMKKENO) and MMKC - version with detailed energy dependence of the cross sections. In MMKK included geometric standard modules close to MMKFK: OOBH – for systems with hexagonal cluster structures and HRAN - for systems with square and rectangular bars. MMKK code provides a detailed multi grouped (299 groups plus sub-groups) calculations of the reactor by the Monte Carlo method. MMKK code is highly effective and at times exceeds the speed of KENO-VI code, which provides a similar design features. In MMKK implemented calculating k-eff perturbations due to changes in material densities and constants. MMKC code has the same geometric standards and uses prepared by NJOY data from files ENDF/B-VII and ROSFOND in a well-known format ACE. As for the shielding calculations as well as for determining diffusion-transport corrections codes KATRIN [12] and DORT-TORT [13] are used. They imply the ABBN data in the ANISN type input format.

For the depletion and kinetic calculations and fuel burn-up codes CARE and ORIGEN are used. They use inlet concentrations of isotopes, neutron flux and irradiation time for counting isotope kinetics of nuclear fuel elements taking into account the nuclear transformations

during radioactive decay and interaction of neutrons causing fission, radiative capture, reactions n,2n, and n,3n. The programs were designed for a wide range of users. A feature of the CARE program [14] is its mobility in use as independently so as a part in a neutronics code (in particular, CARE is used in the TRIGEX complex for burn-up calculations). The ORIGEN program is a part of the SCALE modular system. CARE and ORIGEN codes are both tied completely to the ABBN constants by using modules CONSYST. Analyses of nuclear fuel cycles made by using the CYCLE system code [15] are also carried out on the basis of constants ABBN.

The main feature of all mentioned codes is that – they use one and the same, unique constant's data base like ABBN-93 or ABBN-RF with one and the same code CONSYST for generation of effective cross-sections. The scheme of interrelation of constants and codes used in fast reactor calculations showed in Figure 1.

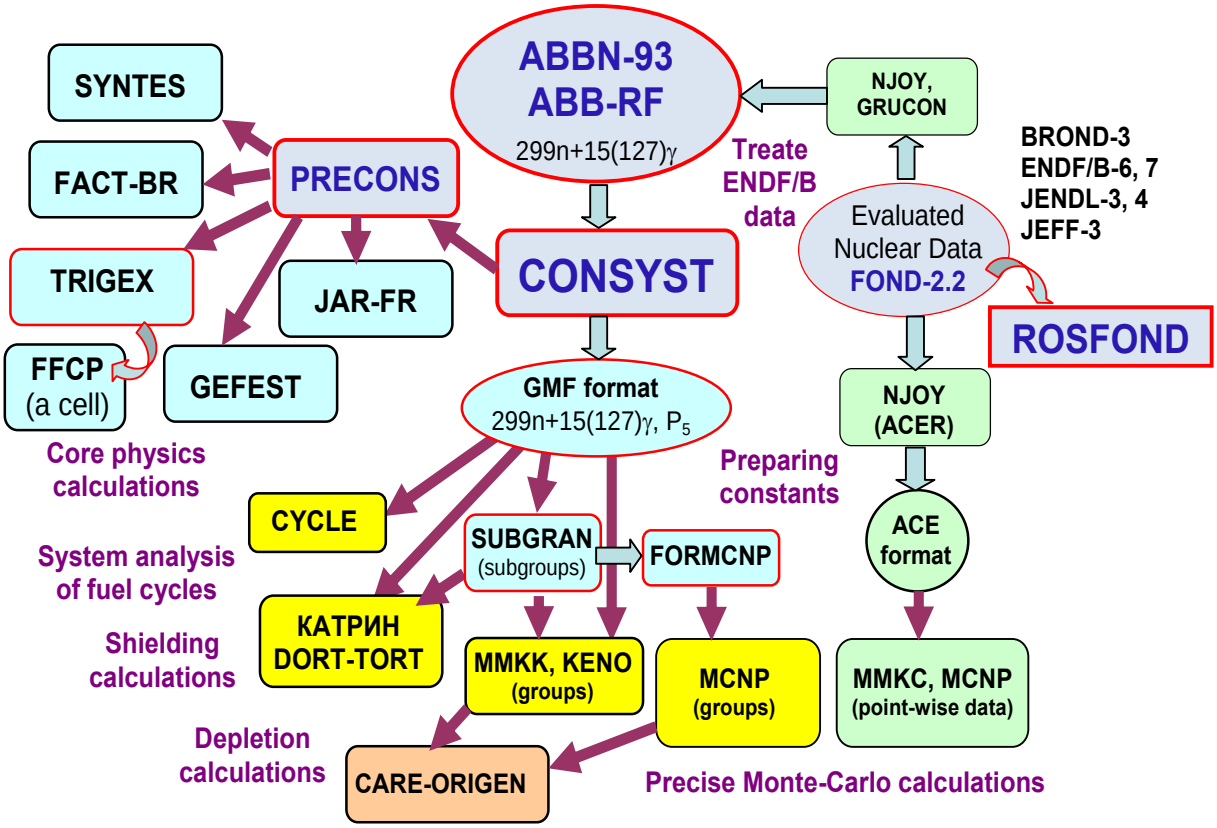

FIG. 1: A scheme of constants and software used in fast reactor neutronics calculations

5. Verification of Codes

The main point in this issue and the hardest one is the task of creation of electronic data bank of comprehensive and full enough experimental information of the "benchmark" type. At this moment we can speak about experiments in nearly hundred different BFS fast critical assemblies in IPPE.

Unfortunately, in many cases this information is too raw and should be re-evaluated in order to be useful for the verification of neutronics codes and nuclear constants. Note that evaluating the accuracy of calculations of the characteristics of fast reactors BN-600 and BN-800 and BN-1200, as well as the perspective of a fast reactor with a lead-cooled BREST-300, are based mainly on the experiments carried out on the BFS. One of the important tasks also

is developing of a methodology for evaluation of the uncertainty of the calculated neutronics parameters.

Nowadays, as a computer system for the accuracy estimation, the INDECS [16,17] data code system is used, see Figure 2.

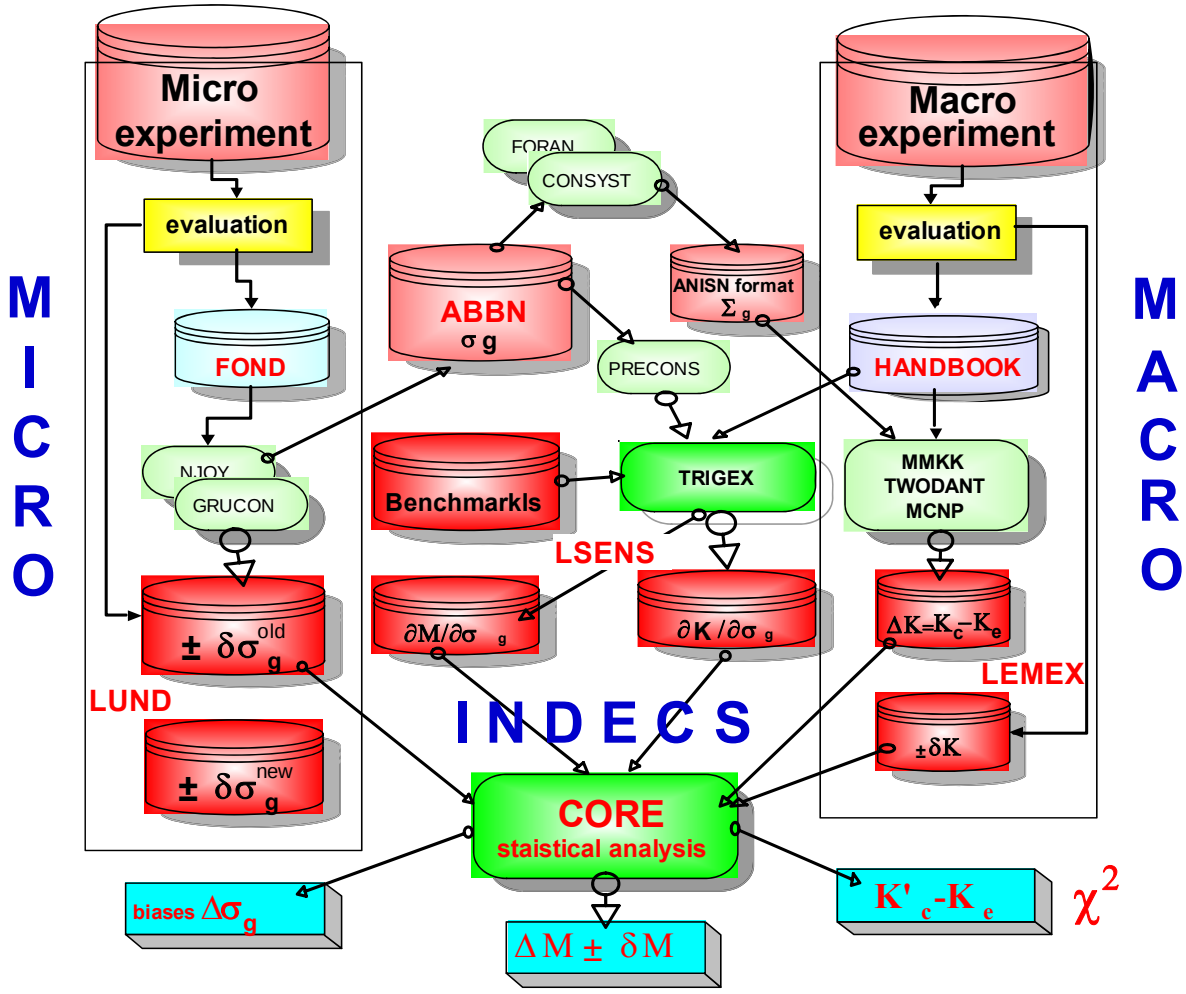

FIG. 2: A scheme of evaluation of calculation uncertainty

The INDECS system includes all the necessary data banks: covariance matrices of group constants (LUND), results of analyses of reactor-physics experiments and their uncertainties (LEMEX), sensitivity coefficients (LSENS), and a statistical code (CORE) for adjustment of constants with use the Maximum Likelihood Method. Currently a technique implemented in the code TRIUM [18] was developing to evaluate technological, methodological and the nuclear data components of error of calculations. The technique is based on GRS method of random drawing multiple sets of input data for the calculation of the physics characteristics of the reactor model (nuclear constants, as well as geometric and material model parameters) using the corresponding error covariance matrices.

6. Summary and Conclusions

Concluding the review of the state of art in the field of constants and software for fast reactors calculations we would like to outline tasks which appear to be solved most important:

o Further improvement of the ROSFOND and ABBN-RF neutron data files based on modern experimental information and scientific knowledge, and introduction data in practice calculations;

o Expansion of experimental data base for validation and verification of computer codes and reactor constants, and development of new benchmarks;

o Further improvement of the error propagation methodology using INDECS and TRIUM code systems, development and updating of the error constant's matrices;

o Further development and introduction of high-precision codes MMK: MMKK - group version and MMKC - with the point-wise cross-sections.

As one of the issues to be resolved, we can give an example, Figure 3 shows not yet unexplained increased divergences between the calculated and experimental data obtained in the assembly BFS-62 in the steel reflector compared with similar data in the uranium blanket. This contradiction between theory and experiment requires an explanation, especially since similar results have been obtained by specialists in France and the United States. Analyses of these results, both in Russia and in the United States, but also in France and Japan, have not yet identified the reasons for these significant differences.

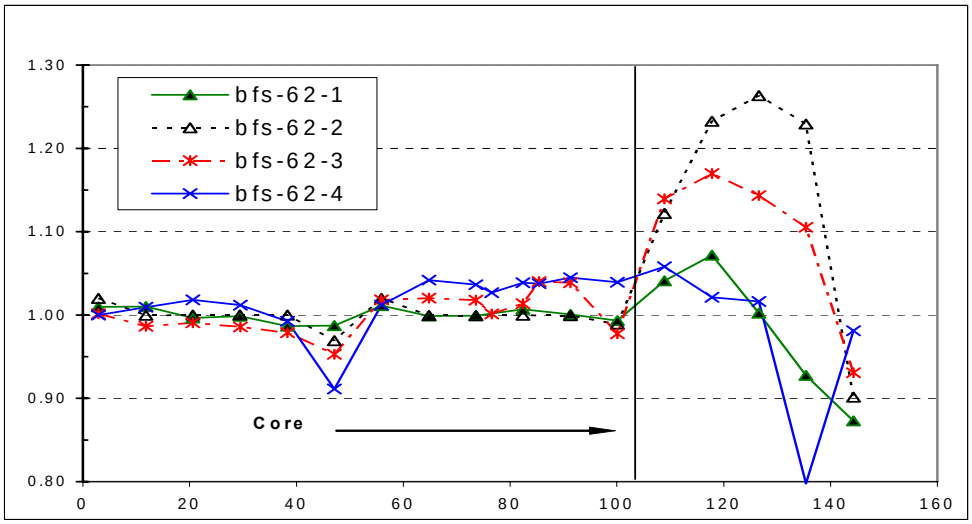

FIG. 3: The discrepancy between reaction rate calculation and experimental results measured in the BFS-62 assembly with steel reflector

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