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Uncertainty Analysis Results on BN-600 Hybrid Core Nuclear Physics Characteristics

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The paper provides prediction uncertainty evaluation results of the nuclear physics parameters of the BN-600 hybrid core. For the evaluation, the analysis results of BFS-62 series of fast reactor benchmark experiments together with other experiments were used. The methodical uncertainty is estimated using IAEA BN-600 test model. The part of uncertainty associated with the nuclear data is evaluated using the adjustment method. Different covariance evaluations are used and their influence is studied.

KEYWORDS: prediction uncertainty, BN-600 hybrid core, steel reflector, BFS-62 benchmark experiments, adjustment method, covariance evaluations.

1. Introduction

One of the practical options to dispose effectively Russian surplus weapons plutonium is the utilization of the BN-600 Russian commercial fast reactor. Since the current BN-600 core consists of uranium dioxide fuel, verification of nuclear characteristics analysis accuracy on mixed-dioxide (MOX) fueled core is required prior to the installation of the MOX fuel into the core. For this goal, in 1999, according to the agreement between JNC and IPPE, a series of critical experiments (BFS-62 benchmark cores, mock-ups for the possible transition of future BN-600 cores) was initiated using the BFS-2 critical facility of IPPE [1]. The experimental analysis was fully finished by the end of 2002 and the results obtained by both JNC and IPPE have been reported [2].

This paper presents the uncertainty evaluation results of the BN-600 hybrid core nuclear physics parameters. They are evaluated by taking into account different kinds of fast reactor core analysis results, including the obtained during the BFS-62 experiment. For the evaluation of the part of uncertainty associated with the nuclear data, the adjustment method is employed, different covariance data are used, and the results are discussed.

2. Analysis Method

For the design development and justification of the BN-600 reactor hybrid core nuclear physics characteristics the code TRIGEX is used in IPPE and JAR in the design organization OKBM. Long time an impediment existed in sure substantiation of the calculation accuracy by using TRIGEX and JAR codes due to a difference in the constants blocks, CONSYST code used in the JAR and PRECON1 used in the TRIGEX. That was the reason for discrepancies in cases when the results expected to be equivalent. To undergo that fault a new program PRECONS was designed and implemented into TRIGEX and JAR codes together with ABBN-93 [3] constant's latest version in parallel, so the dependence on constants in TRIGEX and JAR codes was excluded. Figure 1 shows the sequence of codes and nuclear data used for reactor core calculations. Because the experimental information is mainly formed by BFS experiments, the most of calculations were performed by TRIGEX code, which has a module

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FFCP to allow taking into account the heterogeneous structure of BFS assemblies. FFCP code makes calculations of the cell in subgroup approximation with taking into account one or more resonance nuclides, as a rule ^{235}U, ^{238}U, ^{239}Pu and Fe. To prove the diffusion calculation results and to estimate effects of the cell and assembly heterogeneity as well as the kinetic (transport) effect, the precise calculations are made using Monte-Carlo codes MCNP and MMKKENO [4], and discrete ordinates transport code TWODANT. All the codes TRIGEX, JAR, MMKKENO, and MCNP (in group mode) as well as TWODANT perform calculations on the basis of the common system of group constants ABBN-93.

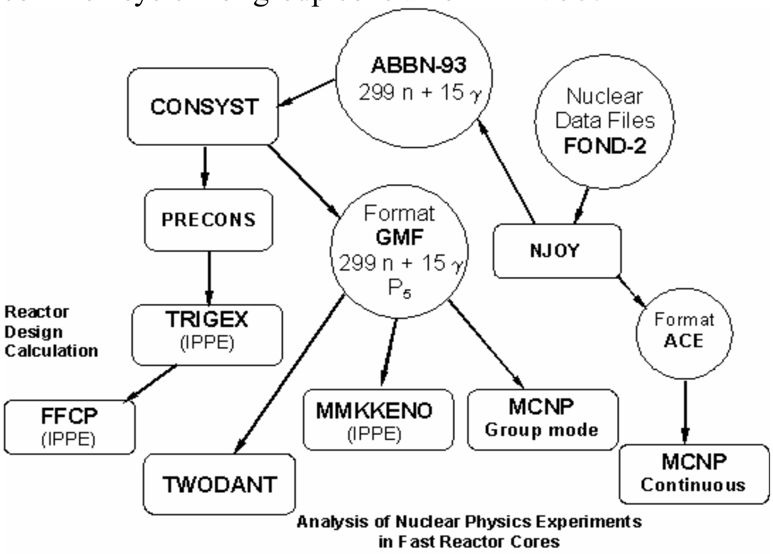

Fig.1 Codes and nuclear data used for in-core nuclear physics calculations

3. BFS-62 Critical Experiment Main Analysis Results

Four main configurations of the BFS-62 critical experiment [1] have been managed in order to obtain measurement data which are important to evaluate possible changes in nuclear characteristics caused by the planned transition of the BN-600 uranium loaded core to the MOX core. Various measurements were made including the criticality, the sodium void reactivity effect, central spectral indices, the fission rate distribution and the control rod worth. Comparison analysis results, obtained by both JNC and IPPE, for all nuclear parameters showed relatively good agreement. Both JNC and IPPE used their own nuclear data libraries and diffusion core calculation codes for the BFS-62 core analysis, which was checked in analyzing JUPITER series data [2]. Table 1 shows the main analysis results.

Table 1 Comparison of experimental and calculated criticality values
[diffusion calculation results corrected on the cell and transport effects]

Assembly	TRIGEX	C/E-1, %	MMKKENO	C/E-1, %
BFS-62-1	0.9976	-0.31	0.9974	-0.33
BFS-62-2	0.9986	-0.23	0.9994	-0.15
BFS-62-3A	1.0002	-0.05	1.0018	+0.11
BFS-62-4	0.9975	-0.34	0.9999	-0.10
ZPPR-9	0.9987	-0.24	0.9985	-0.26

4. Emphasizing of the Steel Reflector Problem

One of the main items of the BN-600 core design is the replacement of uranium blanket by the stainless steel reflector. However, considerable inconsistency between the calculated and measured values (difference of 20% and more) is found for the fission rate distribution outside the core, in the stainless steel reflector region of the cores BFS-62-2 and BFS-62-3A (mock-up with MOX fuel region). Figures 2 and 3 show the C/E analysis results obtained for ^{235}U and ^{239}Pu fission rate distributions in BFS-62 cores.

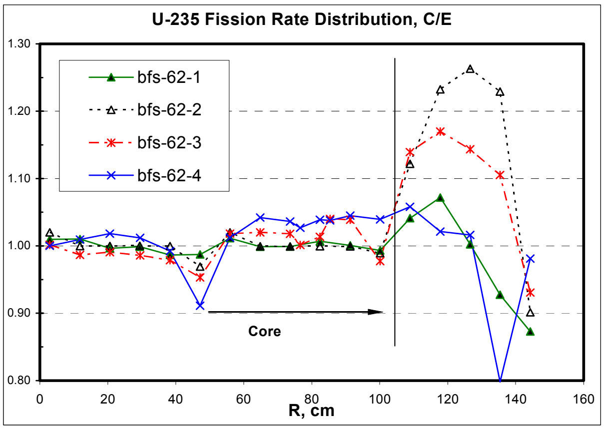

Fig. 2 ^{235}U fission rate distribution

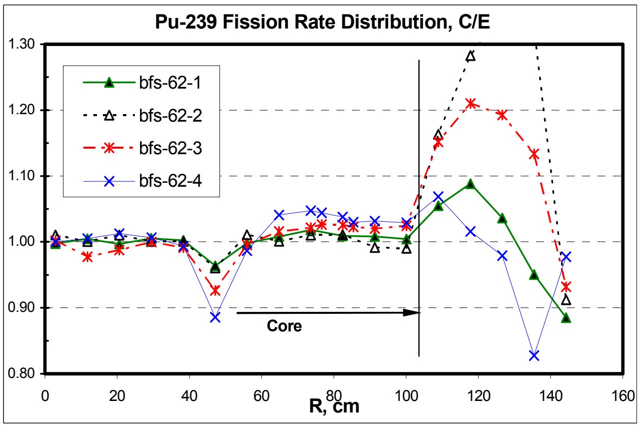

Fig. 3 ^{239}Pu fission rate distribution

5. Total Cross-Section Definition Problem

By the DS_N method the transport of neutrons can be described by the Boltzman steady state equation (1) with the angle independent total cross-section Σ_g^{SN}, as for example made in the code TWODANT:

$$\mu\frac{\partial}{\partial x}\Phi_g(\mu,x)+\Sigma_g^{SN}\ \Phi_g(\mu,x)=\sum_{l=0}^{N}P_l(\mu)\sum_{g'}\Sigma_{sl\,g'\to g}^{SN}\ \Phi_{lg'}+S_g(\mu,x) \qquad (1)$$

$$\Sigma_{sl,g'\to g}^{SN}=\Sigma_{sl}^{g'\to g}\ \text{for}\ g'\neq g\text{, and}\ \Sigma_{sl,g\to g}^{SN}=\Sigma_{sl}^{g\to g}-(\Sigma_{t,l}^{g}-\Sigma_g^{SN})\ \text{for}\ g'=g. \qquad (2)$$

In Monte-Carlo codes (MMKKENO, MCNP) the integral form of the equation is solved

$$\Phi_g(x,\mu)=2\pi\int_0^\infty ds\ e^{-T_g(s)}\left[\sum_{g'}\int_{-1}^{1}d\mu\Sigma_s^{g'\to g}(x-s\mu,\mu'\mu)\Phi_g(x-s,\mu')\,d\mu'+S_g(x-s\mu,\mu)\right]. \qquad (3)$$

Here $T_g(s)=\int_0^s\Sigma_t^g(x-\xi\mu)d\xi$ is the neutron optical length crossing from point *(x-sμ)* to *x*.

The anisotropy of scattering is taken into account by using the P_N approximation and if N is large enough the P_N approximation is quite consistent:

$$\Sigma_{sl}^{g'\to g}(x,\mu)\approx\sum_{l=0}^{N}P_l(\mu)\Sigma_s^{g'\to g}(x). \qquad (4)$$

In ABBN-93 system there are two ways to define the total cross-section Σ_g: 1) current weighted $\Sigma_{t,1}$ and 2) flux weighted $\Sigma_{t,0}$. Rewriting the equation (1) by using these different definitions of the total cross-section Σ_g, we can get two absolutely consistent forms for the Boltzman neutron transport equation:

A: $\Sigma_g^{SN}=\Sigma_{t1}$ (5)

$$\mu\frac{\partial}{\partial x}\Phi_g(\mu,x)+\Sigma_{t1}^{g}\ \Phi_g(\mu,x)=\left[\Sigma_{s,0}^{g'\to g}-(\Sigma_{t,0}^{g}-\Sigma_{t,1}^{g})\right]\Phi_{0g}(x)+\sum_{l=1}^{N}\sum_{g'}P_l(\mu)\sum_{sl}^{g'\to g}\Phi_{l,g'}+S_g(\mu,x)$$

B: $\Sigma_g^{SN}=\Sigma_{t0}$ (6)

$$\mu\frac{\partial}{\partial x}\Phi_g(\mu,x)+\Sigma_{t,0}^{g}(x)\ \Phi_g(\mu,x)=\sum_{s,0}^{g'\to g}\Phi_{0g}(x)+\sum_{l=1}^{N}\sum_{g'}P_l(\mu)\left[\Sigma_{s,l}^{g'\to g}+(\Sigma_{t,0}^{g}-\Sigma_{t,1}^{g})\right]\Phi_{lg}+S_g(\mu,x)$$

It is important to note that:

- If N is finite (as usual, less or equal to 5) than $\Sigma_s^{g\to g'}(x,\mu)$ for some values of μ may be even negative, but it does not influence in the case of the DS_N-method use.
- Form A ($\Sigma_t = \Sigma_{t1}$) sometimes results in negative value of the self-scattering term, that does not influence in case of the DS_N method, but it does not permit in case of the Monte-Carlo code use and should be fixed.
- Form B ($\Sigma_t = \Sigma_{t0}$), even in the isotropic case, requires large N number, and the forms A and B are equivalent in the case of $N \to \infty$ only.
- In absence of resonance structure the forms A and B are equal exactly ($\Sigma_{t1} = \Sigma_{t0}$).
- In practice for description of the albedo from resonance material as iron we have to use the flux-weighted total cross-section $\Sigma_{t,0}$ in the Boltzman equation, but for the consistent description of the leakage term it is better to use $\Sigma_{t,1}$

In the CONSYST code the following 3 parameters control the total cross-section definition: *nmom* controls the number of scattering momentum *l*; *iprib=0/1* defines the form of the total cross-section: 0 – $\Sigma_t = \Sigma_{t0}$; 1 – $\Sigma_t = \Sigma_{t1}$; *idelta=0/1* controls applying the correction $\Delta = \Sigma_{t0} - \Sigma_{t1}$ in the equation (6), 0 means no correction.

6. Analysis Results of the Reactor Physics Benchmarks

To check the uncertainty associated with possible error in definition of the total cross-section within the steel reflector region, a series of experimental benchmark critical configurations with and without reflector from the ICSBEP Handbook [5] were investigated. Table 2 shows calculation results of small metal fueled critical systems specified in [5] which confirmed the made conclusions about the validity of usage$\Sigma_{t,1}$ and $\Sigma_{t,0}$ approaches. The subgroup approach, as expected, gives the most accurate result. Use of the mixed approach for Σ_g, $\Sigma_{t,1}$ in the core and $\Sigma_{t,0}$ in the steel reflector, gives the results close to the subgroup ones.

Table 2 Calculated criticalities for the ICSBEP Handbook benchmark specifications

Identification	Reflector material	Reflector thickness, cm	Total cross-section		Sub-group method
			current-weighted	flux-weighted	
HEU-MET-FAST-014	DeplUran	4.65	0.996	0.996	0.996
HEU-MET-FAST-013	**Iron**	3.65	0.970	**0.987**	**0.997**
HEU-MET-FAST-029	DeplUran	4.7	1.004	1.004	1.004
HEU-MET-FAST-021	**Iron**	9.7	0.964	**0.993**	**1.000**
IEU-MET-FAST-008	DeplUran	3.25	1.006	1.006	1.006
IEU-MET-FAST-005	**Iron**	8.25	0.972	**0.999**	**1.006**
PU-MET-FAST-025	**Iron**	1.55	0.981	**0.990**	**0.997**
PU-MET-FAST-026	**Iron**	11.9	0.966	**0.994**	**1.001**
PU-MET-FAST-028	**Iron**	19.65	0.966	**0.997**	**1.000**
PU-MET-FAST-041	DeplUran	20.98	1.003	1.003	1.003
PU-MET-FAST-032	**Iron**	4.49	0.976	**0.990**	**0.997**
PU-MET-FAST-012	DeplUran	30	1.005	1.005	1.005
PU-MET-FAST-015	**Iron**	30	0.966	**1.006**	**1.006**

Additionally, to check the observed inconsistency within the steel reflector region, a series of critical experiments ZPR-3-53 and ZPR-3-54 [6], and CIRANO ZONA2B experiment [7] performed in MASURCA were involved in the analysis. Those two ZPR-3 assemblies were specially constructed for the purpose to study the steel reflector effect. They had identical core design, loaded with metal uranium-plutonium fuel. In the ZPR-3-53 case the metal uranium was used as the reflector, while the steel was used in the ZPR-3-54. The CIRANO experiment was performed at the MASURCA facility in France for the purpose of investigating the characteristics of plutonium burning fast reactors. During the first phase of the CIRANO experiments three configurations have been investigated in which the fertile breeder blanket was progressively replaced by reflector materials. The experiments were performed in ZONA2A core which has been established in such way that RZ modeling is very representative. The core contained unique PuO_2-UO_2 fuel cells with plutonium enrichment of around 25%. The experiment consists of 3 critical configurations to study the effect of replacement of the uranium fertile blanket by the steel reflector: ZONA2A - fertile blanket around the core (initial state); ZONA2A3 - radial blanket replaced by sodium-steel reflector; ZONA2B - both radial and axial blankets replaced by sodium-steel reflectors.

Table 3 Calculated criticalities by MMKKENO for ZPR-3 and CIRANO cores

Identification	**Reflector material**	**Total cross-section**		**Sub-group method**
		current-weighted	**flux-weighted**	
ZPR-3-53	DeplUran	0.994	0.998	0.998
ZPR-3-54	**Steel**	0.967	**0.997**	**1.002**
ZONA2A	DeplUran	1.001	1.003	1.003
ZONA2A3	**R-Steel**	0.998	**1.006**	**1.005**
ZONA2B	**R&Z-Steel**	0.996	**1.010**	**1.008**

Monte-Carlo MMKKENO code was used for calculating the critical states. Three sets of calculations were done. The first set of calculations performed using the standard procedure, i.e. the current weighted total cross section $\Sigma_{t,1}$ was used as for the core and the reflector. In the second case, mix of $\Sigma_{t,1}$ and $\Sigma_{t,0}$ was used for the core and the reflector, respectively. The third series of calculations performed using the subgroup approach to prepare constants for uranium-238, plutonium-239 and iron. The calculation results obtained are summarized in Table 3. As expected, the results of applying the subgroup approach are the most accurate, and lie more close to the experiment. The use of the mixed approach for the total cross-section, $\Sigma_{t,1}$ for the core and $\Sigma_{t,0}$ for the steel reflector, gives the result which is close to the subgroup approach within 0.2%.

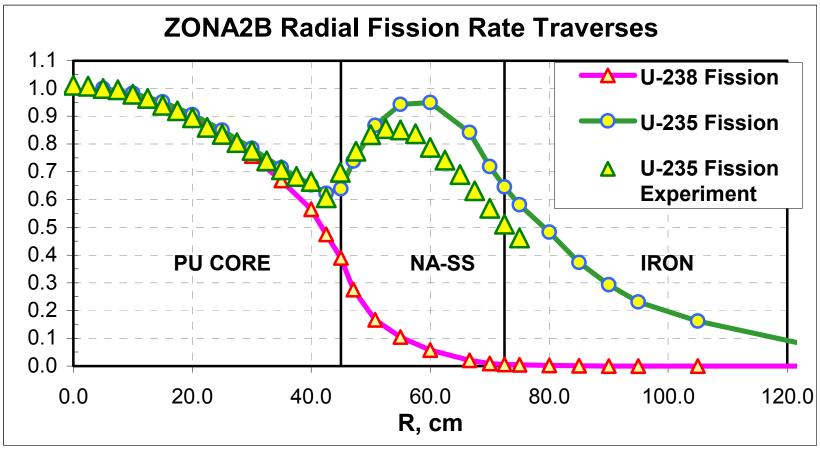

Fig.4 ^{235}U and ^{238}U fission rate distributions in CIRANO ZONA2B core

Figure 4 shows a comparison between calculated and measured uranium-235 and uranium -238 fission reaction rates in the radial direction in the core and in the blanket in CIRANO ZONA2B benchmark core. It is seen the same tendency in discrepancy between calculation and experimental for ^{235}U fission rate distribution, 20% and more, in the stainless steel reflector, same as we saw in BFS-62 cores. Applying the subgroup approach or use the special approach for the total cross-section $\Sigma_{t,1}$ or $\Sigma_{t,0}$ for the steel reflector, does not allow to account for the experimental results. It requires more deep investigations for the problem.

7. Nuclear Data and Methodical Uncertainty Component

Evaluation of the component of the prediction uncertainty associated with nuclear cross section's uncertainty to the main nuclear physics characteristics of the considered variant of the BN-600 hybrid core with the steel reflector is executed. The evaluation is made using the code's and archive's INDECS system [8]. Next BN-600 main physics characteristics were considered: criticalities, efficiency of control and safety rods, sodium void reactivity effect, loss of reactivity by the fuel burn-up. An adjustment procedure using the INDECS system's tools was performed taking into account the experimental analysis results of the investigated benchmark cores, analyzed by using the ABBN-93 nuclear data library. Different scenarios of the adjustment are studied. The main results are presented in Table 4.

Table 4 Main scenarios and results of the adjustment (all the results given in percents)

BN-600 hybrid core nuclear physics parameter	Uncertainty before adjustment	BN-600 uranium core	BFS-62 series	ZPPR-9 & 10	ZPR-3-53 & 54	Other BFS experiments	Adjustment biases	Uncertainty after adjustment
k-eff, %	± 1.5						-0.1	± 0.5
CR, %	± 5.2						+0.8	± 2.2
SR, %	± 6.3						+0.6	± 2.8
SVRE, %Δk/k	± 0.32						-0.07	± 0.19
Burn-up, %Δk/k	± 0.20						+0.03	± 0.15
k-eff, %	± 1.5						+0.1	± 0.3
CR, %	± 5.2						+0.5	± 1.8
SR, %	± 6.3						+0.3	± 1.9
SVRE, %Δk/k	± 0.32						-0.02	± 0.11
Burn-up, %Δk/k	± 0.20						-0.06	± 0.12

The uncertainty "before adjustment" corresponds to so-called "micro-level", it is calculated by using the matrices from LUND library without taking into account any integral or macroscopic reactor experiments. The total uncertainty for k_{eff} is about 1.5%. It should be noted that for the conventional LMFBR core the target criticality calculation accuracy is 0.5% and just this uncertainty can be essentially reduced by taking into account the results of experiments on fast critical assemblies. So, as Table 4 shows, this uncertainty is reduced up to 0.3%. The main sources of k_{eff} uncertainty are caused by uranium, plutonium and iron cross-sections. Table 4 shows the results of two different adjustment scenarios. The accuracy of k_{eff} calculations is significantly reduced from 1.5% to the acceptable size of 0.3-0.5%. The prediction uncertainty for the control rods efficiency is reduced by 2-3 times, for the SVRE is lowered from 0.3 to 0.1%Δk/k, for the loss of reactivity due to the fuel burn-up is reduced from 0.2 to 0.1%Δk/k. It should be noted that the uncertainty of the Doppler effect is practically saved. The considered experiments permit to improve a spectral uncertainty component only which gives small contribution to the total error. Table 4 also shows the BN-600 nuclear parameter biases. They are small and consistent with the estimated uncertainties.

The BN-600 reactor test model described in the IAEA publication [9] is the most convenient for carrying out the comparative analysis of the methodical errors. In the test model full symmetric core arrangement was adopted and two automatic control rods being eliminated. Those and some other simplifications are not significant, so this model in general gives a proper description of the reactor design nuclear physics characteristics. The methodical errors are analyzed by applying different computational methods and codes.

As the result, it was estimated that for the BN-600 reactor with hybrid core, the TRIGEX code provides the calculation uncertainties, as the following:

- k_{eff} value: ±0.6%,
- Worth of shim rods: ±7%; worth of safety rods: ±8%,
- β_{eff} – effective fraction of delayed neutrons: ±6%,
- Sodium void reactivity effect: ±0.003 of Δk/k absolute units,
- Doppler reactivity effect: ±10%,
- Local power rate: ±5% in the core; ±15% in uranium blanket; ±25% in the steel reflector.

8. Nuclear Data Library Effect (NDL Effect)

Independently to the nuclear data adjustment, evaluation of the NDL effect on the criticality of four BFS-62 uranium and plutonium mock-up cores has been done [10]. The NDL effect was estimated as a difference between ABBN-93 and JENDL-3.2 group data, and showed about 0.1%Δk/k difference in BFS-62 uranium fueled cores and 0.3%Δk/k difference in MOX fueled cores. The NDL effect for the BFS-62 MOX cores fully coincide to the k_{eff} uncertainty part associated with nuclear data which was estimated above by the adjustment method taking into account full set of the fast reactor core experiments.

9. Influence of Input Covariance Matrices

To investigate the stability of the adjustment results from priory covariance data used, different variants of covariance matrices ABBN-78 [11], ENDF/B-V and JENDL-3.2 were used. Additionally, own evaluation, called below as a "minimal level", estimated based on calculation of discrepancies between different nuclear data evaluations, is used. The most significant differences are found in the JENDL-3.2 case. For example, for the k_{eff} nuclear data uncertainty component the JENDL-3.2 gives 2 times more optimistic results than ours.

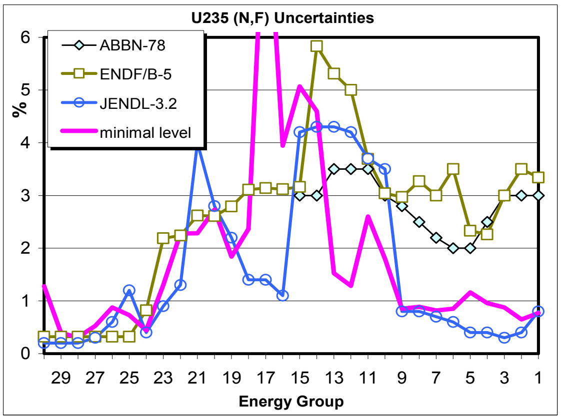

Fig.5 Uncertainties of ^{235}U $\boldsymbol{\sigma_f}$

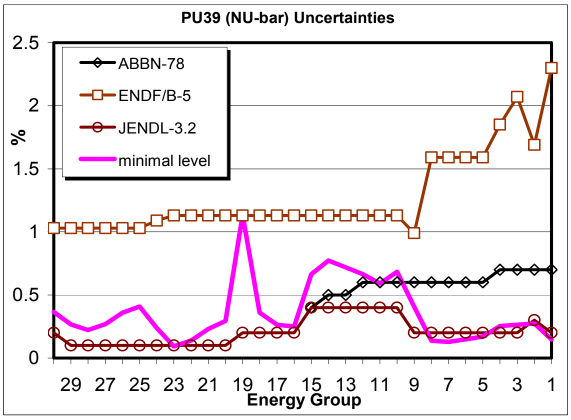

Fig.6 Uncertainties of ^{239}Pu $\boldsymbol{\nu}$ values

Figures 5 and 6 show the comparison of uncertainties adopted in ABBN-78, ENDF/B-V and JENDL-3.2 evaluations for ^{235}U σ_f (similar for ^{239}Pu σ_f) and ^{239}Pu ν value. We can see that for example in case of fission cross-sections ABBN-78 and ENDF/B-V data in the important fast energy region (small group numbers) are very close each other. For the ^{239}Pu ν value they differ 2 times. Nevertheless, JENDL-3.2 evaluations show even the lowest level of the uncertainty in all cases 2-3 times more optimistic. It is interesting that our evaluations, called as a "minimal level" and estimated based on the calculation of discrepancies between different grouped nuclear data, ABBN-78, ENDF/B-V, ENDF/B-VI, JEF-2.2 and JENDL-3.2, lie very close to the JENDL-3.2. The "minimal level" evaluation way lets to estimate a statistical error component mostly and it does not assume, for example, any systematic errors caused by measurement methods. It looks like a difference may exist in the way to include errors of the differential measurements, and a detailed investigation is necessary on the methodology of uncertainty evaluations.

10. Conclusion

The paper presents an essay of the obtained results on the analysis of experiments performed on the Russian facility BFS-2 and some foreign assemblies in substantiation of uncertainties of the possible transition of the BN-600 reactor core. The code TRIGEX was used as the general calculation tool while analyzing the experiments and the design parameters. All the corrections applied were confirmed by the precise Monte-Carlo calculations. The conclusion is made about the prediction uncertainty of the BN-600 hybrid core nuclear physics parameters. The main part of the calculation uncertainty is caused by the nuclear cross-sections inaccuracy. That was estimated by taking into account different kinds of fast reactor core analysis results. A large inconsistency between calculation results and measurements was found in the stainless steel reflector region. It can not be explained by experimental error or by changing treatment approaches. The problem has as a fundamental consequence so it is important for the shielding. So it has to be studied more.

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