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UNCERTAINTY ANALYSIS FOR THE STEADY-STATED BEAVRS BENCHMARK PROBLEM AT ARO AND ARI SITUATIONS

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ABSTRACT

In this paper, the capability of uncertainty propagations of the nuclear-data to the reactor-physics calculations has been implemented in our home-developed code NECP-UNICORN based on the statistical sampling method (SSM). The "two-step" scheme has been applied in NECP-UNICORN to perform the uncertainty analysis for the reactor-physics calculations. For the lattice calculations, the nuclear-data uncertainties are propagated to the few-group constants; then for the core simulations, the uncertainties of the multiplication factor and power distributions introduced by the few-group constants' uncertainties can be quantified. Applying the NECP-UNICORN code, uncertainty analysis has been performed to the BEAVRS benchmark problem at the Hot Zero Power (HZP) conditions, with situations of All Rod In (ARI) and All Rod Out (ARO). From the numerical results, it can be observed that for the multiplication factors of the core simulations, the relative uncertainties are about 5.1‰ for the ARO situation and 5.0‰ for the ARI situation, which are the same magnitude of the relative uncertainties of the fuel assemblies; for the radial power distributions, the relative uncertainties can up to be 4.27% as the maximum value and 2.08% as the RMS value for the ARO situation, and 6.03% as the maximum value and 2.37% as the RMS value for the ARI situation.

1. INTRODUCTION

With the increasing demand for the best-estimate predications to be provided with their confidence bounds in the nuclear research, industry, safety and regulation, the OECD/NEA has organized the UAM ("Uncertainty Analysis in Modeling") expert group to establish the benchmarks for the uncertainty analysis for the coupled multi-physics and multi-scale LWR system [1]. In the reactor system, the reactor-physics calculation is prerequisite for the nuclear safety, reactor design and radiation shielding analysis, which requires the nuclear data as the fundamental input parameters. With the increasing development of the methods, the accuracy of the neutron-physics calculation is mainly limited by the precision of the nuclear data. Moreover, nuclear-data uncertainties exist objectively, as the insufficient measurement precision and the modeling uncertainties. The nuclear-data uncertainties have been proved to be one of the most significant sources of uncertainties for the neutron-physics calculations and received the increasing attentions recently. According to UAM, for the lattice calculations, the uncertainties of the few-group constants should be quantified and for the core simulations, the uncertainties of the important predictions are focused on.

In order to propagate the nuclear-data uncertainties to the important responses of the reactor-physics calculations, the deterministic method and statistical sampling method (SSM) are widely applied. Because of the notable advantages, the SSM has been applied in our-home developed code NECP-UNICORN [2, 3] to perform the uncertainty analysis for the reactor-physics calculations. Based on the "two-step" scheme, the uncertainty-analysis capability for the reactor-physics calculations has been completed in NECP-UNICORN. For the lattice calculations, the nuclear-data uncertainties are propagated to the few-group constants firstly; and then for the core simulations, the few-

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group constants' uncertainties are propagated to the important responses, including the multiplication factor and power distributions. The verifications of NECP-UNICORN have been presented in our previous works [2, 3], and the newly application and researches of NECP-UNICORN for the BEAVRS benchmark problem has been introduced in this paper. Uncertainty analysis has been applied to BEAVRS [4] at the Hot Zero Power (HZP) condition with the situations of All Rod In (ARI) and All Rod Out (ARO). The relative uncertainties of the multiplication factor and power distributions have been quantified and analyzed in detailed for both the ARI and ARO situations.

2. OVERVIEW OF THE NECP-UNICORN CODE

Based on the "two-step" scheme for the reactor-physics calculations, the uncertainty-analysis capability has been implemented in our NECP-UNICORN code. Detailed introductions of the theories and methods have been presented and can be found in our previous works. Therefore, the brief introductions of the capabilities of NECP-UNICORN are focused on in this paper. The brief flowchart of NECP-UNICORN applying SSM to perform the uncertainty analysis is shown in Fig. 1.

For the lattice calculations, a standard multigroup cross-section format has been designed and applied in NECP-UNICORN. This design is due to the fact that for different lattice code, different kinds of multigroup microscopic cross-section library with specific format are required for the lattice calculations. With this standard format, different formatted multigroup microscopic cross-section libraries can be converted into, and hence different lattice code can be added into NECP-UNICORN to carry out the lattice calculations. Until now, DRAGON 5.0 [5] and our home-developed lattice code Bamboo-Lattice [6] (applying the same kernel theories and methods with NECP-CACTI) have been implemented in NECP-UNICORN with the WIMSD-4 [7] and NECL formatted libraries respectively.

As the sampling process is actually the process of perturbations, hence the multigroup cross-section perturbation model has been established and applied in NECP-UNICORN. The relative perturbation factors are generated applying the relative covariance matrices of the multigroup cross-section library. In this multigroup cross-section perturbation model, the actual perturbations of the multigroup cross sections are propagated form the point-wise cross sections, with considering the perturbations of weighting flux due to the perturbation of the cross sections. After the perturbations of the multigroup cross sections, the consistency rules have been implemented to keep the cross sections balance and consistent. Then, the perturbed and consistent multigroup cross sections are converted into the multigroup microscopic cross-section library with the specific format. With the perturbed or samples of the multigroup microscopic cross-section libraries, the lattice calculations are carried out to obtain the samples of the responses. The uncertainty information of the interested responses can be quantified using these responses' samples. The interested responses analyzed for the lattice calculations include the eigenvalue, few-group constants, kinetic parameters and isotope concentrations with depletions.

In the process of uncertainty analysis for the lattice calculations, the samples of the few-group constants can be obtained. Therefore, these samples are directly provided to the uncertainty analysis for the core simulations. This method has the advantage of no requirement to sample for the few-group constants, especially for the core cycle simulations. Our home-developed core code Bamboo-Core [8] has been added into NECP-UNICORN to carry out the core simulations. As the interested responses, the multiplication factor, power distributions, BC curve and AO curve can be analyzed.

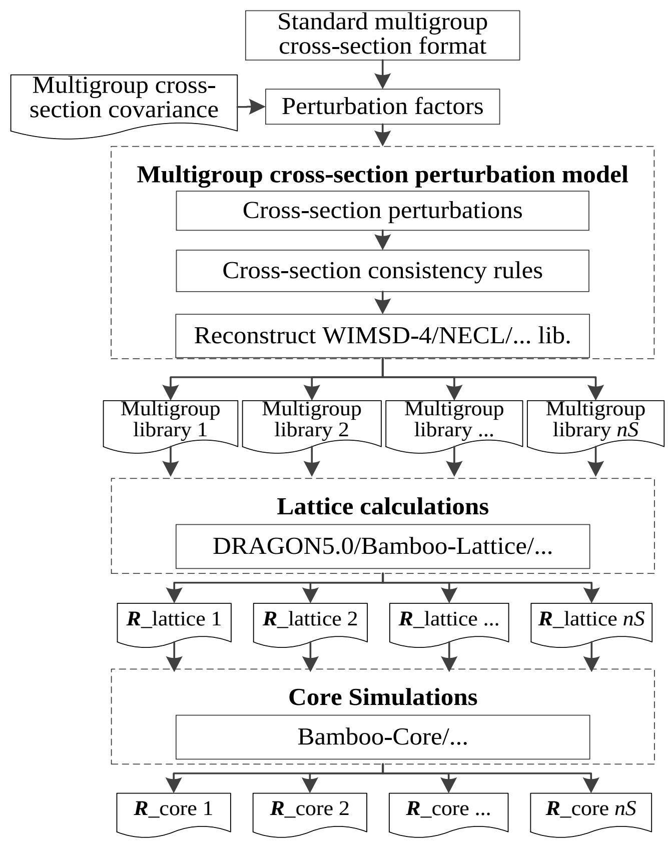

Fig. 1. The brief flowchart of NECP-UNICORN

3. NUMERICAL RESULTS AND ANALYSIS

In this paper, NECP-UNICORN has been applied to the uncertainty analysis for the BEAVRS benchmark problem. The uncertainty analysis has been performed to the HZP condition of BEAVRS with the simulations of ARI and ARO. Our home-developed Bamboo-Lattice and Bamboo-Core are applied to perform the steady-stated modeling and simulation of BEAVRS based on the "two-step" scheme in NECP-UNICORN. For the lattice calculations, the relative uncertainties of the eigenvalue and two-group constants (with cut-off energy for the fast and thermal group set to be 0.625eV) are quantified; for the steady-stated core simulations, the relative uncertainties of the multiplication factor and power distributions are determined.

3.1. Modeling and Simulation of BEAVRS

Verifications of the modeling and simulation with coupled application of Bamboo-Lattice and Bamboo-Core are performed based on the ARO situation. For the ARO situation of BEAVRS at HZP, there are 9 different kinds of fuel assemblies. With the same modeling and simulation parameters given by MIT, including the geometry, temperature, isotope compositions and the configurations, the lattice calculations for these fuel assemblies are modeled and simulated by both Bamboo-Lattice and CASMO-4E [9]. The eigenvalues of these fuel assemblies are compared in Table 1.

Table 1. Eigenvalues of the fuel assemblies of BEAVRS at HZP with ARO

	CASMO-4E	Bamboo-Lattice	Difference/pcm
16000	0.98952	0.98944	-8
24000	1.13130	1.13127	-3
24012	1.00857	1.00837	-20
24016	0.97033	0.97039	6
31000	1.21279	1.21269	-10
31006	1.15630	1.15614	-16
31015	1.07251	1.07253	2
31016	1.05786	1.05793	7
31020	1.02229	1.02256	27

It can be observed that the differences in eigenvalues of the fuel assemblies between Bamboo-Lattice and CASMO-4E are all within 30pcm, which is small and acceptable. These comparisons assure that the modeling and simulations for the fuel assemblies of BEAVRS Bamboo-Lattice are correct. For the core simulation, the multiplication factor obtained by Bamboo-Core is 0.99977 (-23pcm), compared with this by the "one-step" neutron-transport result by CASMO-4E be 1.00031 (+31pcm). The assembly power distributions obtained by Bamboo-Core and CASMO-4E are compared and shown in Fig. 2. The RMS percent difference of the radial power distributions between Bamboo-Core and CASMO-4E is 0.91%. For the "two-step" scheme for the core simulations, these differences are acceptable. Therefore, applying our home-developed Bamboo-Lattice and Bamboo-Core, the correct modeling and simulations of BEAVRS at HZP can be implemented. Moreover, the same modeling methods have been applied to the simulation for the ARI situation, adding the fuel assemblies with the insertion of the control rods.

0.709	0.801	0.804	0.971	0.872	0.966	0.939	1.004
0.699	0.789	0.796	0.960	0.868	0.965	0.948	1.019
-1.4	-1.5	-1.0	-1.1	-0.5	-0.1	1.0	1.5
0.801	0.767	0.938	0.868	1.007	0.900	1.133	1.065
0.789	0.758	0.924	0.862	0.998	0.902	1.135	1.065
-1.5	-1.2	-1.5	-0.7	-0.9	0.2	0.2	0.0
0.804	0.938	0.864	1.022	0.913	1.010	0.941	0.939
0.796	0.924	0.858	1.012	0.912	1.010	0.948	0.955
-1.0	-1.5	-0.7	-1.0	-0.1	0.0	0.7	1.7
0.971	0.868	1.022	0.951	1.095	1.024	1.187	0.779
0.960	0.862	1.012	0.951	1.096	1.031	1.187	0.776
-1.1	-0.7	-1.0	0.0	0.1	0.7	0.0	-0.4
0.872	1.007	0.913	1.095	1.444	1.193	1.269	
0.868	0.998	0.912	1.096	1.442	1.207	1.262	
-0.5	-0.9	-0.1	0.1	-0.1	1.2	-0.6	
0.966	0.900	1.010	1.024	1.193	1.250	0.936	
0.965	0.902	1.010	1.031	1.207	1.280	0.936	
-0.1	0.2	0.0	0.7	1.2	2.4	0.0	
0.939	1.133	0.941	1.187	1.269	0.936		
0.948	1.135	0.948	1.187	1.262	0.936		
1.0	0.2	0.7	0.0	-0.6	0.0		
1.004	1.065	0.939	0.779			CASMO-4E	
1.019	1.065	0.955	0.776			Bamboo-Lattice+Core	
1.5	0.0	1.7	-0.4			Difference/%	

Fig. 2. The comparison of the assembly power distributions

Based on the modeling and simulations for BEAVRS using Bamboo-Lattice and Bamboo-Core, uncertainty analysis has been performed at HZP with the condition of ARO and ARI. For the ARI situation, the critical boron concentration is set to be 686pcm according to the operation data of BEAVRS.

3.2. Uncertainty Results for the Lattice Calculations

For the lattice calculations, the relative uncertainties of the eigenvalue and two-group constants have been quantified. The nuclides and corresponding cross-section types analyzed in the uncertainty analysis are listed in Table 2. The uncertainty-analysis results of the fuel assemblies are shown in Table 3 for ARO situation and Table 4 for ARI situation.

Table 2. The nuclides and cross-section types analyzed

Cross section	Nuclides analyzed
$\sigma_{(n,elas)}$	^{234}U,^{235}U,^{238}U,^{1}H,^{16}O,^{90}Zr,^{91}Zr,^{92}Zr,^{10}B,^{11}B
$\sigma_{(n,inel)}$	^{234}U,^{235}U,^{238}U,^{90}Zr,^{91}Zr,^{92}Zr,^{10}B,^{11}B
$\sigma_{(n,2n)}$	^{234}U,^{235}U,^{238}U,^{90}Zr,^{91}Zr,^{92}Zr
σ_f	^{234}U,^{235}U,^{238}U
σ_γ	^{234}U,^{235}U,^{238}U,^{1}H,^{16}O,^{90}Zr,^{91}Zr,^{92}Zr,^{10}B,^{11}B
ν	^{235}U,^{238}U
σ_α	^{16}O,^{10}B,^{11}B

It can be observed that the relative uncertainties for the eigenvalues of the fuel assemblies vary from 5.0‰ to 5.8‰; and the largest relative uncertainties of the two-group constants can up to be 1.70% for D_1, the fast-group diffusion coefficient. Moreover, the relative uncertainties of the fast-group constants are larger than those of the thermal group, and the response uncertainties at ARO situation almost have the same magnitude with those at ARI situation.

3.3. Uncertainty Results for the Core Simulations

Provided with the samples of the two-group constants for the fuel assemblies at the ARO and ARI situations, uncertainty analysis have been performed to the steady-state core simulations. The relative uncertainties of the multiplication factor of BEAVRS at HZP are shown in Table 5.

Table 5. Relative uncertainties of the multiplication factor

Situation	*keff*	$\Delta k_{eff}/k_{eff}$/%
ARO	0.99977	0.51
ARI	0.99921	0.50

It can be observed that the relative uncertainties of the multiplication factors are 5.1‰ for the ARO situation and 5.0‰ for the ARI situation, with the same magnitude of the relative uncertainties of the eigenvalues for the fuel assemblies.

The relative uncertainties of the power distributions in both ARO and ARI situations are compared and shown in Fig. 3.

0.699	0.789	0.796	0.960	0.868	0.965	0.948	1.019
4.27	3.98	3.47	2.68	1.91	0.70	0.66	1.69
0.789	0.758	0.924	0.862	0.998	0.902	1.135	1.065
3.98	3.82	3.21	2.60	1.66	0.62	0.89	1.79
0.796	0.924	0.858	1.012	0.912	1.010	0.948	0.955
3.47	3.21	2.80	2.02	1.27	0.16	0.99	1.87
0.960	0.862	1.012	0.951	1.096	1.031	1.187	0.776
2.68	2.60	2.02	1.30	0.28	0.59	1.63	1.93
0.868	0.998	0.912	1.096	1.442	1.207	1.262	
1.91	1.66	1.27	0.28	0.83	1.55	2.23	
0.965	0.902	1.010	1.031	1.207	1.280	0.936	
0.70	0.62	0.16	0.59	1.55	2.28	2.52	
0.948	1.135	0.948	1.187	1.262	0.936		
0.66	0.89	0.99	1.63	2.23	2.52		
1.019	1.065	0.955	0.776			Assembly Power	
1.69	1.79	1.87	1.93			Rel. Unc./%	

(a). For ARO situation

0.176	0.381	0.297	0.980	1.090	0.981	0.416	0.851
6.03	5.23	4.39	2.79	2.31	1.95	1.33	0.64
0.381	0.436	0.623	0.916	1.240	0.969	0.877	0.914
5.23	4.78	3.81	2.71	2.03	1.68	0.65	0.84
0.297	0.623	0.376	1.050	1.120	1.070	0.476	0.910
4.39	3.81	3.58	2.22	1.59	0.84	0.42	1.39
0.980	0.916	1.050	1.030	1.140	1.200	1.340	0.882
2.79	2.71	2.22	1.71	0.78	0.65	1.88	2.04
1.090	1.240	1.120	1.140	0.787	1.430	1.790	
2.31	2.03	1.59	0.78	0.43	2.12	2.99	
0.981	0.969	1.070	1.200	1.430	1.780	1.450	
1.95	1.68	0.84	0.65	2.12	3.16	3.44	
0.416	0.877	0.476	1.340	1.790	1.450		
1.33	0.65	0.42	1.88	2.99	3.44		
0.851	0.914	0.910	0.882			Assembly Power	
0.64	0.84	1.39	2.04			Rel. Unc./%	

(b). For ARI situation

Fig. 3. The relative uncertainties of the power distributions

It can be observed that the relative uncertainties of the power distributions at ARI are larger than those at ARO. For the situation of ARO, the maximum relative uncertainty is 4.27% occurred in the middle assembly, and the RMS value of the relative uncertainties is 2.08%; for the situation of ARI, the maximum relative uncertainty is 6.03% occurred in the middle assembly, and the RMS value of the relative uncertainties is 2.37%.

From the view of the reactor-physics calculations, the uncertainties of the multiplication factor and power distributions introduced by the nuclear-data uncertainties are notable. Moreover, the uncertainties are expected higher for the depleted core at HFP than those for the fresh-fueled core at HZP. Therefore, these nuclear-data uncertainties should be taken into account for the safety analysis and economic competitiveness of the reactor system.

4. CONCLUSIONS

In this paper, the uncertainty-analysis capability for the reactor-physics calculations based on the "two-step" scheme has been implemented in our home-developed NECP-UNICORN code. The nuclear-data uncertainties are firstly propagated to the important responses of the lattice calculations, including the eigenvalue, few-group constants, kinetic parameters and so on; and then to the significant responses of the core simulations, including the multiplication factor, power distributions and so on. With NECP-UNICORN, the uncertainty analysis has been performed to the BEAVRS benchmark problem at the HZP condition with the ARO and ARI situations. The relative uncertainties of the eigenvalue and few-group constants for the lattice calculations and multiplication factor and power distributions for the steady-state core simulations have been quantified. Notable uncertainties can be observed for the important responses of the reactor-physics calculations for the fresh-fueled core, these uncertainties will be higher for the depleted core. Therefore, the uncertainty analysis will be focused on the cycle calculations and transient calculations in the further researches.

Table 3. Uncertainty-analysis results with ARO situation

	k_∞	D_1	D_2	$\Sigma_{a,1}$	$\Sigma_{a,2}$	$\nu\Sigma_{f,1}$	$\nu\Sigma_{f,2}$	$\Sigma_{s,1,1}$	$\Sigma_{s,1,2}$	$\Sigma_{s,2,1}$	$\Sigma_{s,2,2}$
16000	0.57	1.65	0.37	1.02	0.44	1.03	0.39	1.01	1.18	0.57	0.35
24000	0.52	1.62	0.37	0.96	0.39	0.76	0.38	1.01	1.11	0.55	0.35
24012	0.52	1.64	0.37	0.96	0.34	0.76	0.38	1.01	1.18	0.53	0.36
24016	0.52	1.65	0.37	0.96	0.33	0.75	0.38	1.02	1.21	0.53	0.36
31000	0.50	1.60	0.37	0.92	0.36	0.64	0.38	1.00	1.08	0.54	0.35
31006	0.50	1.61	0.37	0.92	0.34	0.63	0.38	1.00	1.11	0.54	0.36
31015	0.50	1.62	0.37	0.92	0.32	0.63	0.38	1.01	1.16	0.52	0.36
31016	0.50	1.63	0.37	0.93	0.31	0.63	0.38	1.01	1.16	0.52	0.36
31020	0.50	1.63	0.37	0.93	0.30	0.62	0.38	1.01	1.18	0.52	0.36

Table 4. Uncertainty-analysis results with ARI situation

	k_∞	D_1	D_2	$\Sigma_{a,1}$	$\Sigma_{a,2}$	$v\Sigma_{f,1}$	$v\Sigma_{f,2}$	$\Sigma_{s,1,1}$	$\Sigma_{s,1,2}$	$\Sigma_{s,2,1}$	$\Sigma_{s,2,2}$
16000	0.58	1.64	0.37	1.02	0.46	1.03	0.39	1.01	1.15	0.59	0.35
24000	0.53	1.61	0.37	0.96	0.41	0.76	0.38	1.00	1.09	0.56	0.35
24012	0.52	1.63	0.37	0.96	0.36	0.76	0.38	1.01	1.16	0.54	0.36
24016	0.52	1.64	0.37	0.96	0.34	0.76	0.38	1.01	1.19	0.53	0.36
31000	0.51	1.59	0.37	0.92	0.37	0.64	0.38	1.00	1.06	0.55	0.35
31006	0.50	1.60	0.37	0.92	0.36	0.64	0.38	1.00	1.09	0.54	0.36
31015	0.50	1.62	0.37	0.92	0.33	0.63	0.38	1.01	1.14	0.53	0.36
31016	0.50	1.62	0.37	0.92	0.32	0.63	0.38	1.01	1.15	0.53	0.36
31020	0.50	1.63	0.37	0.93	0.31	0.63	0.38	1.01	1.17	0.52	0.36
16000R	0.57	1.70	0.36	1.05	0.31	1.08	0.39	1.02	1.49	0.51	0.36
24000R	0.53	1.66	0.36	0.98	0.29	0.77	0.38	1.00	1.38	0.50	0.36

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