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Sensitivity Analysis Of PWR Keff To Multi-temperature Nuclear Data By RMC

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Abstract –*Monte Carlo code is traditional verification methods for nuclear power software. It should have high precision coupled with burnup and thermal feedback, and nuclear data is still one factors affecting the accuracy of calculation. Sensitivity analysis of core keff to multi-temperature nuclear data will be done in this paper for Monte Carlo simulation couple with burnup. First the basic process of thermal feedback calculation for Monte Carlo code is introduced. Then the sensitivity coefficients of PWR keff to multi-temperature nuclear data is calculated under the given temperature and nuclide with simplified model for studying the effect of multi-temperature nuclear data separately. It is found that the variation of sensitivity coefficient of keff to major nuclear data vs. temperature increases with the fuel depletion, and the change trend is different between ^{235}U with ^{239}Pu. At last, the demand for multi-temperature nuclear data in Monte Carlo simulation with thermal feedback is proposed, especially for simulating a PWR with MOX fuel or a deeper burnup.*

I. INTRODUCTION

In order to simulate actual reactor operating conditions, Monte Carlo code should calculate core with thermal feedback. Thermal feedback is also necessary for Monte Carlo code to be a verification or design software.

The basic process of thermal feedback calculation for Monte Carlo code is shown in Fig. 1. When the moderator and fuel temperature and density is updated, new cross sections for Monte Carlo transport calculation can be generated from interpolation of multi-temperature nuclear data which generated by Doppler broadening[1] .

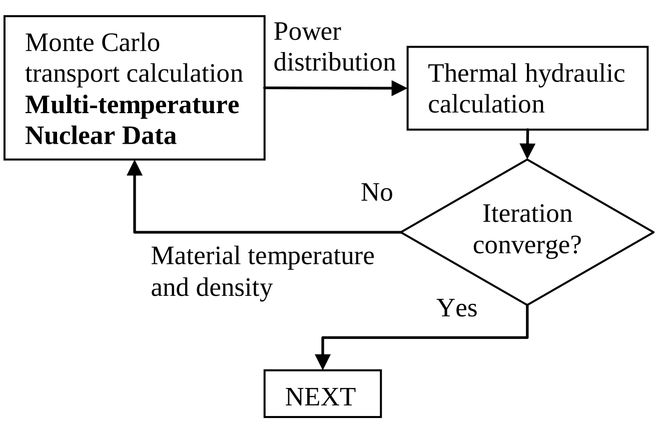

Fig. 1 Flow Chart of Monte Carlo with Thermal Feedback

As shown above, at least two issues need to be considered in thermal feedback calculation, one problem is how to realize and accelerate the convergence of transport and thermal coupling calculation in complex and fine grid, Another problem is to consider the influence of multi-temperature nuclear data on the calculation accuracy.

This paper attempts to establish a path to study the influence of multi-temperature nuclear data on the core calculation precision through the sensitivity analysis, and proposes the demand for multi-temperature nuclear data in Monte Carlo simulation with thermal feedback.

II. DESCRIPTION OF THE ACTUAL WORK

Multi-temperature nuclear data in this paper refers to the continuous energy cross section library for Monte Carlo transport calculation. Because this paper takes PWR as the calculation object, so the material mainly refers to uranium fuel and water. In order to eliminate the interference of the density to calculated results, only the material temperature is taken into account in the sensitivity analysis.

Because Monte Carlo transport calculation coupled with thermal feedback for whole PWR core is very time-consuming and factors affecting the accuracy would be complex, so the sensitivity coefficients of PWR keff to multi-temperature nuclear data is calculated under the given temperature and nuclide with simplified model for studying the effect of multi-temperature nuclear data separately.

The multi temperature continuous energy cross section library used in this paper is the CENACE library[2] which is developed based on CENDL Evaluation Database by China Nuclear Data Center, with temperature from 293K to 2200K, fully covering the existing PWR temperature range. However, due to the lack of covariance data in the CENACE library[3], only the sensitivity coefficients are calculated based on the library.

The Monte Carlo code used in this paper is RMC[4] which is developed by the engineering physics department of Tsinghua University, with the sensitivity analysis function of PWR keff To Nuclear Data .

Based on CENACE and RMC, sensitivity analysis have been done for typical PWR in China. The sensitivity coefficients of PWR's keff to multi-temperature nuclear data are calculated in this paper. The basic framework of sensitivity and uncertainty analysis based on chinese nuclear data and Monte Carlo code will be initially formed, then further work can be carried out within this framework.

1. Description of the sensitivity analysis method [4-5]

Solving keff to the sensitivity coefficient of nuclear data is an important step to analyze uncertainty. Sensitivity coefficient is defined as

$$S^k_{x^p_{i,g}} = \frac{dk/k}{dx^p_{i,g} / x^p_{i,g}}. \qquad (1)$$

Where $x^p_{i,g}$ is the nuclear data in i reaction type and g group of nuclide p . As we know, the sensitivity coefficient $S^k_{x^p_{i,g}}$ describe the relative change of keff caused by the relative change of nuclear data $x^p_{i,g}$.

The representational software is the 1d sensitivity and uncertainty analysis sequence(TSUNAMI-1D) and 3d sensitivity and uncertainty analysis sequence (TSUNAMI-3D) which is developed by the Oak Ridge National Laboratory(ORNL). They have the same based on a group of cross section database, and need to perform a calculation of uniting colonies and resonance self-shielding, therefore must calculate the implicit sensitivity coefficient for considering the influence to calculate the cross section by the method of resonance self-shielding. They get the flux and accompanying flux though doing a calculation of forward and accompanying transport.

In order to make the calculation process more simple and the results more reliable, how to calculate the sensitivity coefficient of keff to nuclear data by the continuous-energy Monte Carlo Code becomes one of the hot topics in the study of Monte Carlo method in recent years. Iterated Fission Probability(IFP) is almost adopted by sensitivity analysis continuous-energy Monte Carlo Code， such as MCNP6、McCARD、SERPENT2 et, owing to the clear physical concept and high calculation precision. According to the physical meaning of the accompanying flux, IFP directly get the estimated value of accompanying flux in forward transport process, rather than to perform an extra calculation of accompanying transport. The Monte Carlo Code RMC which is developed by the engineering physics dept of Tsinghua University already has the function that analyzes the sensitivity coefficient of nuclear data based on the keff of IFP.

2 Qin ShanⅡ first cycle's core model

Qin ShanⅡ is the 2nd generation nuclear power plant with China's first independent design. Two 6500000 kW pressurized water reactor are used. Typical low enrichment fuel and loading pattern are used in the reactor core. Therefore, the analysis results for Qin ShanⅡ first cycle's core in this paper may apply to the 2nd+ generation nuclear power plant and even the 3rd generation nuclear power plant.

Qin ShanⅡ first cycle's core Core is made up of 121 AFA3G fuel assemblies[6]. Each assembly has 264 fuel rods, 24 guide tubes which can place the control rods, burnable poison rods or neutron source and 1 instrument tube. The height of core with active is 365.8cm, equivalent diameter is 267.0cm, thermal power is 1930MWt.

Fuel rods are made by low enriched UO_2 pellet in the Zr-4 alloy tube, fuel rods also filled with pressurized helium. The control rod guide thimble and gauge pipe are made up of zirconium alloy.

Main design parameters are shown in table 1, core model is shown in figure 2.

Table 1 Main design parameters of core

The parameters of fuel rod (cold state)		The parameters of fuel assembly (cold state)	
Outside diameter of cladding，cm	0.95	grid number	17×17
Inside diameter of cladding，cm	0.836	rod center distance，cm	1.26
Cladding material	Zr-4	pellet center distance，cm	21.504
The filled gas	helium	The number of fuel rod	264
UO2 pellet diameter，cm	0.819	number of guide tube	24
The volume proportion	0.9882	number of instrument	1
The active height ，cm	365.8	Assembly number	121

A total of nine computational cases are involved in the model, including the combination of three temperature points and three burnup steps. The material cross section (including thermal scattering data) of 293K, 600K, 900K are chosen, which coverage the normal operation condition of PWR. In order to analyze the conditions of different fuel depletion, the BOL, MOL and EOL steps of the core life are selected, and the calculation results of the nuclide composition are derived from the official design software of the reactor.

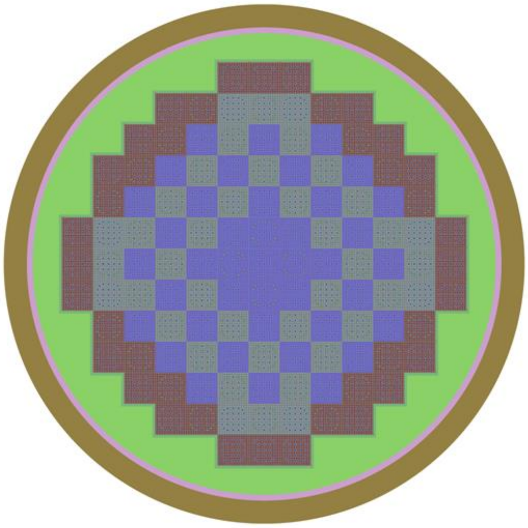

Fig. 2 Loading pattern of first cycle

III. RESULTS

Each case involves sensitivity coefficients of more than one hundred kinds of nuclides, Including hundreds of calculations of the core keff with the change of the nuclear cross section. This paper lists several important nuclides cross section's results for each case, include first 12 nuclear cross sections according to the absolute value of the sensitivity coefficient in descending order, shown in Table 2, table 3 and table 4.

As shown in the table, the sensitivity coefficient of keff to the same nuclear cross section will change with temperature:

a) some will reduce, such as sensitivity coefficient of keff to ^{235}U average fission neutron number in Table 2, some will increase, such as sensitivity coefficient of keff to ^{239}Pu average fission neutron number in Table 4.

b) At the beginning of life (BOL) the sensitivity coefficient's variation of keff to major nuclear data between 293K with 900K is about 3%;

c) With the fuel depletion, the content of ^{239}Pu in the core also increase, so the influence of the ^{239}Pu cross section on the core accuracy is gradually improved.

d）At the end of life (EOL) the sensitivity coefficient's variation of keff to major nuclear data between 293K with 900K is about 10%.

There are two interesting phenomena：1) with the fuel depletion, the variation of sensitivity coefficient of keff to major nuclear data vs. temperature increases, 2) the change trend of sensitivity coefficient of keff to ^{235}U average fission neutron number vs. temperature is different from sensitivity coefficient of keff to ^{239}Pu average fission neutron number.

Then the multi-group sensitivity coefficients of keff to ^{235}U/^{239}Pu average fission neutron number are calculated, shown in Figure 3 and figure 4.

Table 2 Sensitivity coefficients and RSD in BOL

Nuclear data	Reactor type	The sensitivity coefficient			Relative Standard Deviation (RSD)		
		293 K	600 K	900 K	293 K	600 K	900 K
^{235}U	average fission neutron number	9.335E-01	9.330E-01	9.323E-01	2.439E-05	2.488E-05	2.538E-05
^{235}U	total fission	4.412E-01	4.441E-01	4.531E-01	1.090E-04	1.099E-04	1.059E-04
^{235}U	total cross section	3.417E-01	3.451E-01	3.543E-01	1.758E-04	1.789E-04	1.743E-04
^{238}U	absorb	-1.908E-01	-1.939E-01	-1.962E-01	1.113E-04	1.075E-04	1.151E-04
^{10}B	total cross section	-1.655E-01	-1.651E-01	-1.645E-01	1.240E-04	1.178E-04	1.297E-04
^{10}B	absorb	-1.655E-01	-1.651E-01	-1.645E-01	1.238E-04	1.168E-04	1.279E-04
^{1}H	elastic scattering	1.438E-01	1.545E-01	1.613E-01	2.301E-03	2.143E-03	2.092E-03
^{238}U	total cross section	-1.418E-01	-1.430E-01	-1.434E-01	1.067E-03	1.071E-03	1.128E-03
^{1}H	total cross section	9.258E-02	1.004E-01	1.075E-01	5.073E-03	4.630E-03	4.428E-03
^{235}U	absorb	-9.946E-02	-9.893E-02	-9.878E-02	9.384E-05	8.992E-05	9.532E-05
^{1}H	absorb	-7.134E-02	-7.086E-02	-6.933E-02	1.056E-04	9.881E-05	1.031E-04
^{238}U	average fission neutron number	6.647E-02	6.702E-02	6.775E-02	3.426E-04	3.463E-04	3.493E-04

Table 3 Sensitivity coefficients and RSD in MOL

Nuclear data	Reactor type	The sensitivity coefficient			Relative Standard Deviation (RSD)		
		293 K	600 K	900 K	293 K	600 K	900 K
^{235}U	average fission neutron number	6.462E-01	6.406E-01	5.944E-01	7.421E-05	6.891E-05	8.138E-05
^{235}U	total fission	3.241E-01	3.231E-01	3.076E-01	1.731E-04	1.580E-04	1.440E-04
^{239}Pu	average fission neutron number	2.586E-01	2.634E-01	3.009E-01	1.747E-04	1.567E-04	1.740E-04
^{235}U	total cross section	2.575E-01	2.572E-01	2.384E-01	2.479E-04	2.349E-04	2.452E-04
^{238}U	absorb	-1.838E-01	-1.864E-01	-1.860E-01	1.103E-04	1.158E-04	1.151E-04
^{239}Pu	total fission	1.501E-01	1.532E-01	1.776E-01	2.983E-04	2.773E-04	2.496E-04
^{10}B	total cross section	-1.450E-01	-1.439E-01	-1.392E-01	1.394E-04	1.364E-04	1.377E-04
^{10}B	absorb	-1.450E-01	-1.439E-01	-1.392E-01	1.372E-04	1.366E-04	1.360E-04
^{1}H	elastic scattering	1.383E-01	1.488E-01	1.530E-01	2.277E-03	2.238E-03	2.121E-03
^{238}U	total cross section	-1.294E-01	-1.306E-01	-1.298E-01	1.222E-03	1.203E-03	1.227E-03
^{239}Pu	total cross section	9.519E-02	9.664E-02	1.050E-01	4.866E-04	4.680E-04	4.506E-04
^{1}H	total cross section	9.465E-02	1.027E-01	1.139E-01	4.557E-03	4.305E-03	4.078E-03

Table 4 Sensitivity coefficients and RSD in EOL

Nuclear data	Reactor type	The sensitivity coefficient			Relative Standard Deviation (RSD)		
		293 K	600 K	900 K	293 K	600 K	900 K
^{235}U	average fission neutron number	4.957E-01	4.887E-01	4.396E-01	9.837E-05	1.052E-04	1.098E-04
^{239}Pu	average fission neutron number	3.671E-01	3.727E-01	4.195E-01	1.296E-04	1.341E-04	1.105E-04
^{235}U	total fission	2.593E-01	2.567E-01	2.445E-01	1.997E-04	2.098E-04	1.995E-04
^{239}Pu	total fission	2.172E-01	2.208E-01	2.292E-01	2.300E-04	2.302E-04	2.221E-04
^{235}U	total cross section	2.103E-01	2.085E-01	1.843E-01	2.736E-04	2.792E-04	3.080E-04
^{238}U	absorb	-1.845E-01	-1.870E-01	-1.852E-01	1.194E-04	1.200E-04	1.200E-04
^{10}B	total cross section	-1.450E-01	-1.437E-01	-1.374E-01	1.489E-04	1.474E-04	1.445E-04
^{10}B	absorb	-1.450E-01	-1.437E-01	-1.374E-01	1.482E-04	1.463E-04	1.416E-04
^{239}Pu	total cross section	1.418E-01	1.432E-01	1.472E-01	3.783E-04	3.800E-04	3.720E-04
^{1}H	elastic scattering	1.323E-01	1.414E-01	1.475E-01	2.526E-03	2.328E-03	2.099E-03
^{238}U	total cross section	-1.280E-01	-1.293E-01	-1.278E-01	1.279E-03	1.204E-03	1.210E-03
^{1}H	total cross section	9.309E-02	9.990E-02	1.177E-01	5.104E-03	4.763E-03	3.837E-03

As shown in the figure 3 and figure 4, the variation of multi-group sensitivity coefficients is mainly concentrated in the thermal groups. Because the core energy spectrum hardening with the fuel depletion, the thermal neutron's value increases, so the total variation of multi-group sensitivity coefficients vs. temperature will increase.

Because the average fission neutron number of nuclide varies with the energy, and the change trend is different between ^{235}U with ^{239}Pu, as shown in figure 5. So the change trend of sensitivity coefficient of keff to ^{235}U average fission neutron number vs. temperature is different from sensitivity coefficient of keff to ^{239}Pu average fission neutron number.

Temperature rise will lead to further hardening of the core energy spectrum, which is conducive to ^{239}Pu fission but not conducive to ^{235}U fission. Therefore, the sensitivity coefficient of keff to ^{239}Pu average fission neutron number increases with temperature rise, and the sensitivity coefficient of keff to ^{235}U average fission neutron number decreases with temperature rise.

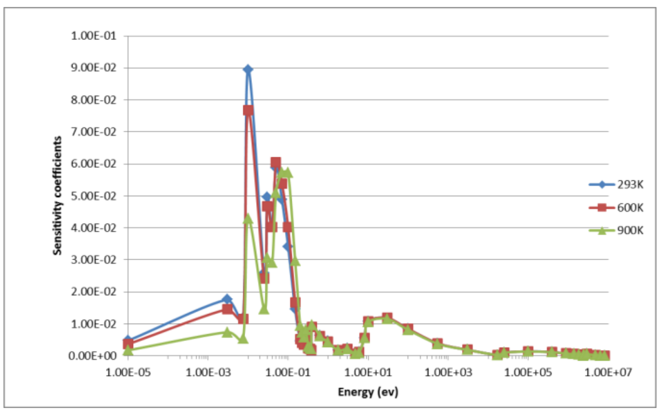

Fig. 3 Sensitivity coefficients of k_{eff} to ^{235}U average fission neutron number (EOL)

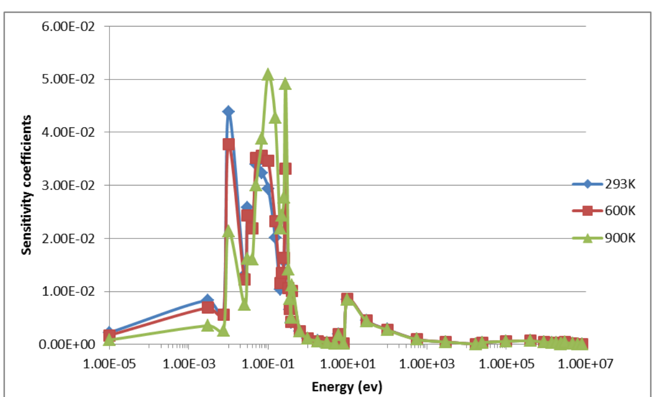

Fig. 4 Sensitivity coefficients of k_{eff} to ^{239}Pu average fission neutron number (EOL)

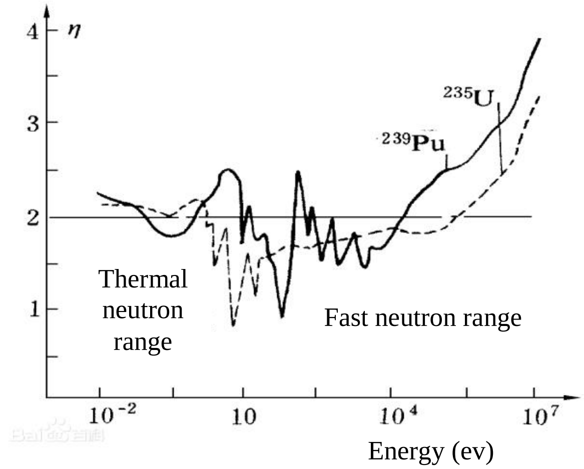

Fig. 5 ^{235}U/^{239}Pu average fission neutron number

IV. CONCLUSIONS

In this paper, the sensitivity analysis of PWR keff to the multi-temperature nuclear data CENACE used in Monte Carlo calculation with thermal feedback have been done by RMC. It is found that the variation of sensitivity coefficient of keff to major nuclear data vs. temperature increases with the fuel depletion, and the change trend is different between ^{235}U with ^{239}Pu.

Temperature rise will lead to larger sensitivity coefficient of core keff to ^{239}Pu cross section. This trend may lead to an increase in the uncertainty of the core keff for a PWR with MOX fuel or a deeper burnup.

Therefore, when simulate similar core by Monte Carlo core with thermal feedback, it may be necessary to reduce uncertainty caused by multi-temperature nuclear data such as ^{239}Pu nuclear data, such as to obtain a more accurate temperature nuclear data and covariance data etc.

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