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NAPTH: Neutronics analysis code system for the fusion–fission hybrid reactor with pressure tube type blanket

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HIGHLIGHTS

- A new code system for the neutronics analysis of the fusion–fission hybrid reactor with pressure tube type blanket is developed.
- The code system can perform the neutrions analysis based on precise geometry.
- The numerical results indicate that the code system is reliable and efficient for the conceptual design of a pressure tube type fusion–fission hybrid reactor.

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ABSTRACT

The fusion–fission hybrid reactor is considered as a potential path to the early application of fusion energy. A new concept with pressure tube type blanket has recently been proposed for a feasible hybrid reactor. In this paper, a code system for the neutronics analysis of the pressure tube type hybrid reactor is developed based on the two-step calculation scheme: the few-group homogeneous constant calculation and the full blanket calculation. The few-group homogeneous constants are calculated using the lattice code DRAGON4. The blanket transport calculation is performed by the multigroup Monte Carlo code. A link procedure for fitting the cross sections is developed between these two steps. An additional procedure is developed to calculate the burnup, power distribution, energy multiplication factor, tritium breeding ratio and neutron multiplication factor. From some numerical results, it is found that the code system NAPTH is reliable and exhibits good calculation efficiency. It can be used for the conceptual design of the pressure tube type hybrid reactor with precise geometry.

1. Introduction

At present, light water reactors (LWR) and CANDU reactors are the most widely used commercial reactors. The fuel management technology and the cooling technology of fission reactors have been well developed in the past decades. However, the fission reactors are faced with some difficulties: the shortage of uranium resources, the high-level waste, the criticality safety issue and nuclear proliferation. Although nuclear fusion has an enormous potential to provide a safe, clean and unlimited energy source, it is generally recognized that the commercial pure fusion reactors will not be run in the short period.

If the fusion–fission hybrid reactor (FFHR) can be constructed in terms of the mature fission technologies and the current or nearly expected plasma and fusion technologies, it will make a very promising way for the early application of fusion energy. The FFHR needs lower plasma conditions than those in a pure fusion reactor. The blanket is driven by high-energy fusion neutrons, so it can be employed for nuclear waste incineration, fissile fuel breeding and energy production. The nuclear fuel can multiply neutrons, which makes tritium self-sufficiency easily to be achieved. A FFHR is operated in a sub-critical state without the criticality safety issue. Therefore, the investigation of the FFHR has positively been carried out worldwide [1–6].

In the previous studies on the FFHR, there are two kinds of tools for the neutronics analysis, namely, continuous energy Monte Carlo (MC) codes and deterministic transport codes. The Monte Carlo MC codes are famous for their capability of simulating continuous energy and complex geometry problem. The transport-burnup codes consisting of continuous energy MC code MCNP and burnup calculation code ORIGEN2 are widely used [6–9]. On the other hand, the deterministic transport codes with good computation efficiency, such as ANISN, SCALE and BISON based on the S_N method, have also been utilized for the neutronics analysis of the FFHR [10–12]. The neutronics analysis code system VisualBUS, which uses either the MC method or S_N method for transport calculation, has been employed for the designs of FDS series fusion power plants [4].

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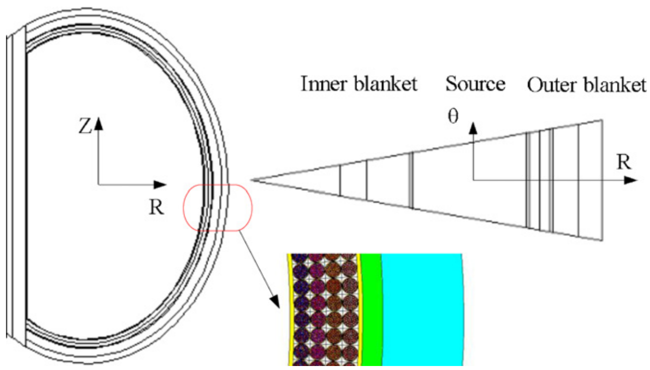

Fig. 1. Cross sections of the calculation model.

The author's research group has proposed a feasible pressure tube type blanket for energy production [13]. In the blanket, the pressurized light water is contained in the pressure tubes to avoid the pressure from acting on the first wall and structure wall directly. It is helpful to decrease the thickness of the first wall and keep the property of source neutrons [13]. The cross-section of the pressure tube type FFHR is shown in Fig. 1. The ITER-scale tokamak is applied as a fusion neutron source. Because there are 20 toroidal coils in ITER, the torus is modeled by 20 equal sectors [1,14] as shown in Fig. 1. The plasma chamber is encompassed by inner blanket and outer blanket. The inner blanket is only designed for tritium breeding. The outer blanket consists of the first wall, fission zone, structure wall, tritium breeding zone and reflect layer. The cross sectional view of the outer blanket is presented in Fig. 2. There are rows of pressure tube bundles in the outer blanket. The pressure tube bundles are put along the poloidal direction, namely perpendicular to the radial direction and the axial direction. In Ref. [13], the neutronics calculations have showed that it is feasible for the pressure tube type FFHR to be constructed based on the ITER level plasma and fusion technologies which are much lower than those in a pure fusion plant. This condition is very effective to keep intactness of the plasma facing material. The pressure tube type FFHR can burn discharged PWR spent fuel without the partition of heavy isotopes, which is helpful for exploiting the uranium resource efficiently.

In Ref. [13], the evaluation was carried out by a continuous energy MC code based on a simplified model with layered homogeneous material. However, the code is not so suitable for the evaluation of different design with real geometry, since the efficiency is not satisfying. In the proposed pressure tube type FFHR, there are tens of thousands of fuel pins in each sector of the blanket. The structure is complex. If the neutronics calculations are directly carried out by continuous energy MC code, it will require very long computation time to tally the information for burnup calculations and evaluation of the performances of the FFHR. When there are many fissile isotopes and the sub-criticality level is low, it is more time-consuming to get converged results. Besides, although the deterministic transport codes have good computation efficiency, they can hardly perform the neutronics calculations, due to its poor performance in the description of such a complex geometry. So a code of high calculation efficiency and high calculation precision is needed to perform the neutronics design based on real geometry.

In this paper, a new code named NAPTH (Neutronics Analysis for the Pressure Tube type Hybrid reactor) is developed using a two-step calculation scheme which has been widely and successfully used in the fission reactor design, namely the lattice few-group homogeneous constant calculation and the full blanket calculation. This code can efficiently perform the neutronics analysis of the pressure tube type hybrid reactor with precise geometry.

The material composition and structure of the pressure tube bundles in the proposed FFHR are similar to those in CANDU reactors. The code DRAGON has been widely applied for the lattice calculations in the neutronics design and analysis of CANDU reactor [15]. The code of DRAGON Version 4 (DRAGON4) [16] containing many advanced neutronics methods can provide accurate lattice constants. Because there are many different typical assembly states to be calculated, the code DRAGON4 based on deterministic algorithms shows much better computation efficiency than continuous energy MC code. For these reasons, the lattice calculations are performed by the code DRAGON4 in this paper.

The homogenized blanket is a 3D 'D-shape' model with a toroidal segment of 18°. In the blanket, there are some different zones including the fission zone, the tritium breeding zone and the reflect layer. The complex geometry is difficult for the deterministic code to describe it. The MC code has a powerful capability of handling complex geometries, so the blanket transport calculation is done by the multigroup MC code [17]. In this paper, only the flux in the homogenized assemblies is to be tallied by the multigroup MC code, so the computation efficiency will be acceptable.

The calculation methods applied in NAPTH are introduced in Section 2. The code is tested with a 17×17 PWR assembly problem and the IAEA ADS benchmark in Section 3. In Section 4, the code system is used for the neutronics analysis of a pressure tube type FFHR. In the final section, some conclusions are given.

2. Calculation methods and code system

A code named NAPTH has been developed applying the two-step calculation scheme: the lattice few-group homogeneous constants calculation and the blanket calculation. The computational flow chart is presented in Fig. 3.

2.1. Lattice few-group constants calculations

In this paper, the lattice calculations are performed by the code DRAGON4 involving many accurate neutronics methods [16]. In this code, the resonance self-shielding calculations can be performed by the subgroup projection method in association with the optimized SHEM-361 energy group structure [18,19]; the types of transport correction and the maximum level of anisotropy for the scattering cross sections are arbitrarily selected according to the problem; a number of different algorithms for solution of the neutron transport equation such as the collision probability method and the method of characteristics are contained. The code DRAGON4 with these methods is sufficient for the lattice calculations.

In order to cover different blanket cases, we select some typical assembly cases with different burnups, power levels and coolant densities for the lattice calculations. Then the condensed and

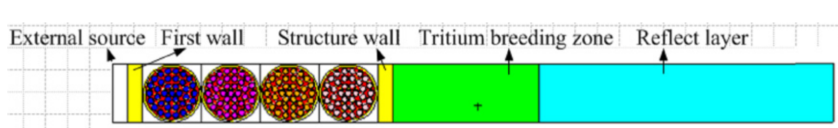

Fig. 2. Cross sectional view of the outer blanket.

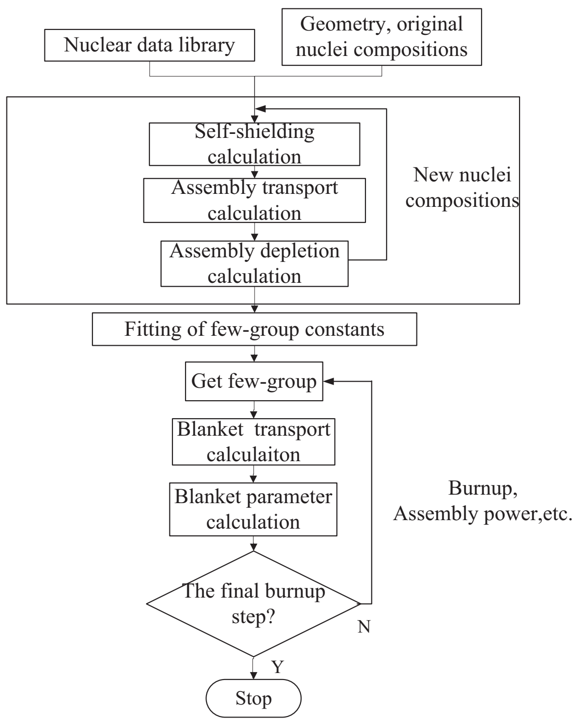

Fig. 3. The computational flow chart of NAPTH.

homogenized few-group parameters of the assemblies are obtained by the homogenization method of flux-volume weighting [20].

2.2. Fitting of few-group constants

A procedure for fitting the few-group cross sections is developed to link the two steps. In this procedure, the few-group macroscopic cross sections obtained from the lattice calculations are fitted to a function of different variables mentioned above. It is well known for the neutronics calculations of fission reactor that the reactivity is greatly affected by the neutron absorption of poison isotopes. So the few-group macroscopic cross sections must take into account the effect of poison isotopes such as ^{135}Xe and ^{149}Sm. The few-group macroscopic cross sections are finally written as follows:

$$\Sigma_l = f_{1l}(\mathrm{BU}, U)\cdot f_{2l}(P_r) + k\times(N_{\mathrm{Xe}}\sigma_{\mathrm{Xe}}(\mathrm{BU}) + N_{\mathrm{Sm}}\sigma_{\mathrm{Sm}}(\mathrm{BU}))\quad \begin{cases} k=1,\ l=t,a \\ k=0,\ l\neq t,a \end{cases} \tag{1}$$

where, Σ_l ($l=t, s, a, f, \ldots$) are respectively the macroscopic total cross section, the scattering cross section, the absorption cross section and the fission cross section, etc. $f_{il}(x)$ ($x=1.2$) is a polynomial of variable x, obtained by the least square method. BU is the burnup. P_r is the relative power. U is the coolant density which is obtained by thermal–hydraulic feedback. σ_{Xe} and σ_{Sm} are the microscopic absorption cross sections of ^{135}Xe and ^{149}Sm, respectively. N_{Xe} and N_{Sm} are the atomic densities of ^{135}Xe and ^{149}Sm.

2.3. Blanket transport calculation

The blanket transport calculation is done by the multigroup MC code, and the flux in the homogenized assemblies is tallied by the multigroup MC code. The values of the flux are used to calculate the burnup and some important parameters such as neutron multiplication factor (K_s), energy multiplication factor (M) and tritium breeding ratio (TBR) as described in the next section. Because only the flux in the homogenized assemblies is to be tallied, the calculation time will be reduced to an acceptable level.

2.4. Blanket parameter calculation

After the blanket transport calculation, another procedure is developed. In this procedure, the burnups, relative powers of assemblies and the atomic densities of poison isotopes ^{135}Xe and ^{149}Sm are calculated to get the few-group constants. Moreover, in the design of a hybrid reactor, some important parameters such as neutron multiplication factor (K_s), energy multiplication factor (M) and tritium breeding ratio (TBR) are also evaluated in this procedure. The calculation methods of these important parameters are discussed as follows.

Because of the existence of external neutron source, the neutron multiplication factor K_s should be calculated as below:

$$K_s = \frac{\int_V\int_E \nu\Sigma_f\cdot\phi\, dE\, dV}{\int_V\int_E \nu\Sigma_f\cdot\phi\, dE\, dV + S} \tag{2}$$

where S is the intensity of external neutron source; υ is the average number of neutrons released per fission.

In the hybrid reactor, fission reactions have a significant contribution to the energy multiplication, although there exits other exothermic nuclear reaction. In this paper, M is defined as:

$$M = \frac{\int_V\int_E \kappa\Sigma_f\cdot\phi\, dE\, dV}{E_{\mathrm{source}}} \tag{3}$$

where E_{source} is the energy of external neutron source.

Tritium can be extracted from the breeding reactions of ^{6}Li and ^{7}Li isotopes, as is given below:

$$^6\mathrm{Li} + n \to \alpha + T + 4.784\,\mathrm{MeV} \tag{4}$$

$$^7\mathrm{Li} + n \to \alpha + T + n' - 2.467\,\mathrm{MeV} \tag{5}$$

So TBR is calculated as follows:

$$\mathrm{TBR} = \int_V\int_{E(n,T)} (\Sigma^{6\mathrm{Li}} + \Sigma(n,T,n')^7\mathrm{Li})\cdot\phi\, dE\, dV \tag{6}$$

where $\Sigma(n,T)^6$Li and $\Sigma(n,T,n')^7$Li are the tritium breeding macroscopic cross sections of ^{6}Li and ^{7}Li, respectively.

3. Numerical verification

The calculation methods described above have been implemented in the code NAPTH. This code is verified by a 17×17 PWR assembly problem and the IAEA ADS benchmark.

3.1. PWR assembly problem

To verify the accuracy of the code, a 17×17 PWR assembly problem is computed. The geometry of the assembly is shown in Fig. 4. The pitch is 1.26 cm. The enrichment of ^{235}U is 3.1%. The 1/8 assembly as shown in Fig. 4 is calculated. The reference results are obtained by WIMS9 [21], DRAGON4 and HELIOS [22]. The numerical results are shown in Fig. 5. At the beginning of the lifetime (BOL), the relative errors of the K_{eff} compared with the reference results are 120 pcm, 30 pcm and 20 pcm, respectively, while they are 1200 pcm, 90 pcm and 230 pcm at the end of the lifetime (EOL). NAPTH has high calculation accuracy.

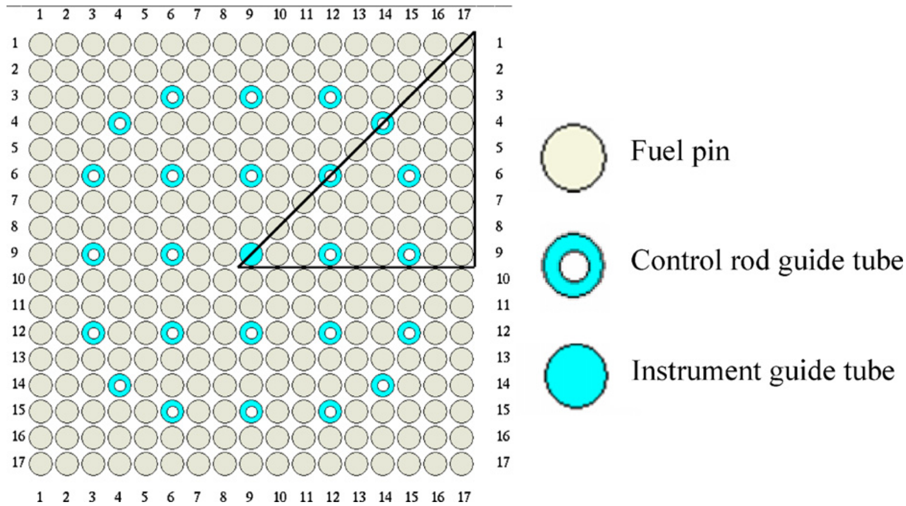

Fig. 4. Cross section of the PWR assembly.

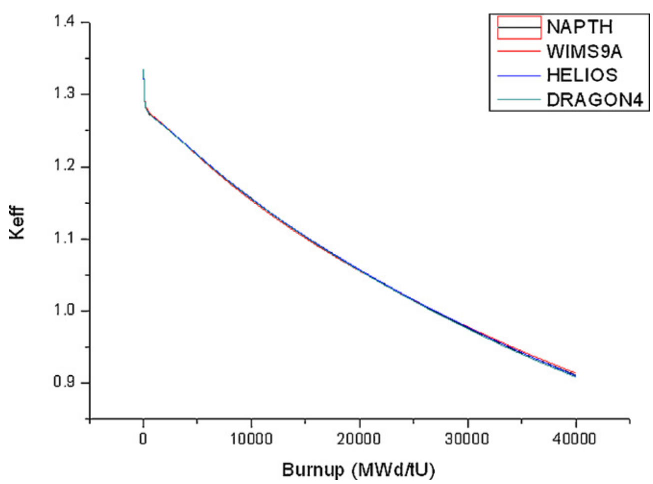

Fig. 5. K_{eff} curve as a function of burnup.

3.2. IAEA ADS benchmark

The IAEA ADS benchmark issued by CRP is a ^{233}U–^{232}Th fuelled cylindrical symmetry system. The details of the benchmark and principal neutronic features can be found in Ref. [23]. In this paper, NAPTH is applied to calculate the benchmark. The three fuel zones are divided into 19 spatial mesh intervals in the radial direction and 10 in the axial direction. A WIMSD–IAEA 172-group library [24] obtained from JEFF-3.1 is used. Considering the space, the numerical results of the situation, where initial K_{eff} is equal to 0.96 are shown. The results include initial enrichments of ^{233}U, spallation source effectiveness φ^*, the evolution curves of K_{eff}, the evolution curves of external source and the spatial distributions of power density at BOL. All these results have a good agreement with those of other participants.

Results of the enrichment are presented in Table 1. The difference among the results lies in the different neutron cross-section libraries and the different calculation methods. Spallation effectiveness is rather high in the benchmark. The results at BOL are shown in Table 2. In the benchmark, the core is fuelled with tons of fertile material (^{232}Th) and fissile material (^{233}U). Variety of the net mass of fissile material and accumulation of fission products make K_{eff} varying during the lifetime. Fig. 6 depicts the K_{eff} evolution results. The source intensity should be adjusted to maintain the required total power (1500MWt) during fuel burnup. The source evolution results are presented in following Fig. 7. Since a sufficiently strong external source presents in ADS, the spatial distribution of power density behaves very differently from a critical system. The radial distributions and the axial distributions are shown in Figs. 8 and 9.

Table 1
Initial enrichments of ^{233}U (in %).

Participants	Russia (MC)	Italy	Japan	France	German	NAPTH
ε	9.925	9.96	9.4	9.94	9.68	9.77

Table 2
Spallation source effectiveness.

Participants	Russia	Italy	Sweden	France	Belarus	NAPTH
φ^*	1.33	1.31	1.24	1.27	1.25	1.329

4. Calculations of the pressure tube type FFHR

In this section, NAPTH is employed for the neutronics analysis of a pressure tube type FFHR. The calculation model is a sector as

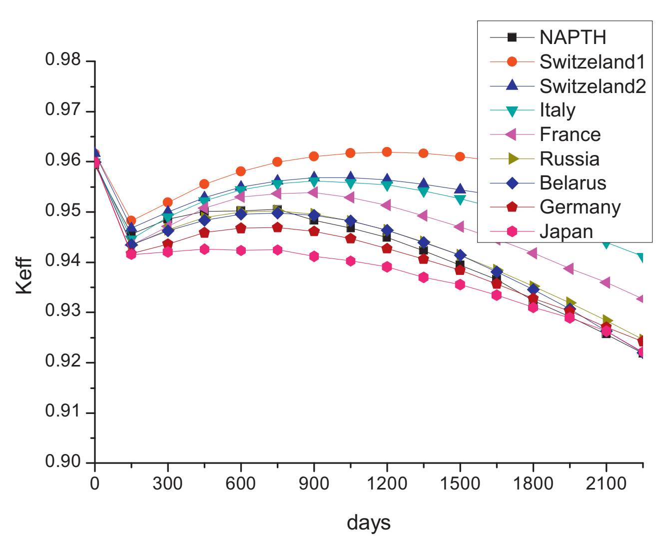

Fig. 6. The evolution of K_{eff} during fuel burnup.

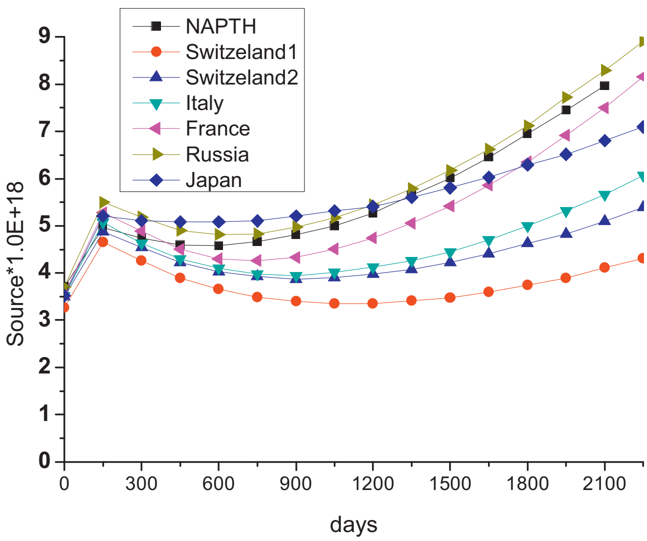

Fig. 7. The evolution of source intensity during fuel burnup.

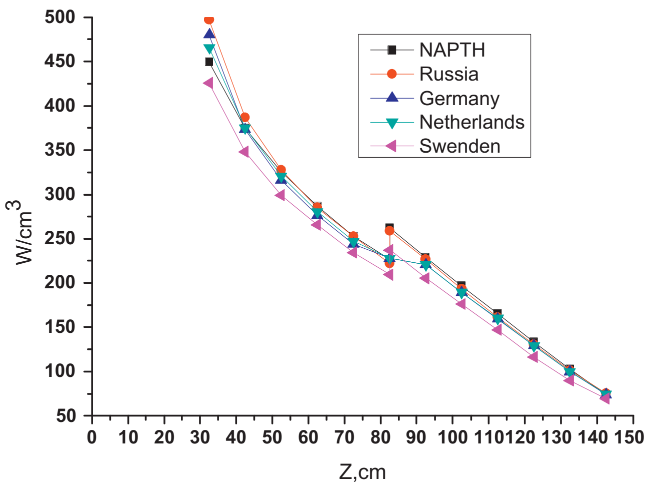

Fig. 8. Radial power distribution at $z = 0$ cm.

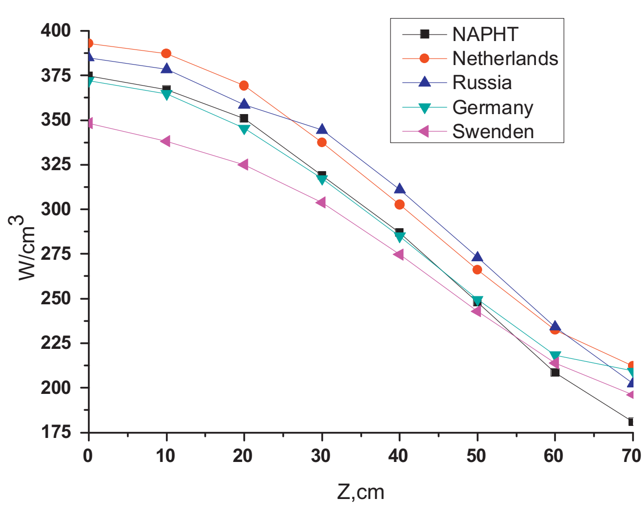

Fig. 9. Axial power distribution at $R = 42.5$ cm.

Table 3
Main geometry parameters of the FFHR.

Parameter (cm)	
Major radius	620.0
Minor radius	200.0
First wall	2.0
Fuel zone	32.0
Tritium breeding zone	10.0
Reflect layer	40.0

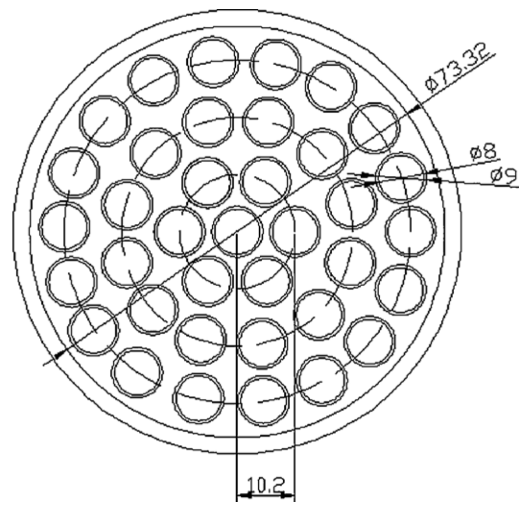

Fig. 10. Structure of the pressure tube (mm).

depicted in Fig. 1. The geometry parameters of the FFHR are given in Table 3. In each sector, there are four rows of pressure tube bundles surrounding the fusion neutron source, and 159 bundles are loaded in each row. In each bundle, there are 37 fuel pins, and the total number of the fuel pins in each sector is 23532. The structure of a bundle is given in Fig. 10.

The material composition is shown in Table 4. The stainless steel (SS316) is used as the first wall material and the structure material. Natural uranium oxide is selected as fission fuel, and light water is utilized as coolant in the assembly. The solid Li_2O is selected as the tritium breeder, and the enrichment of ^6Li is 90%. Graphite is used as the reflector. Helium is filled in the gaps of the tubes.

In the calculations, the blanket keeps 150 MW fission power output for 300 days subdivided into 18 burnup steps. A SHEM-361 group format library derived from ENDF/B-VII is used for the neutronics calculations. In the lattice calculations, the subgroup method is selected for the resonance self-shielding calculation; the APOLLO type transport correction is used; the method of characteristics is applied for the transport calculation. In the FFHR, because the source neutrons come into the system from one direction, the angular flux distribution shows significant anisotropic. The approximation of reflective boundary condition made in the lattice calculation will cause large errors in the lattice constants. To take into account the effect of fusion neutrons, the lattice calculations are carried out on a supercell as shown in Fig. 2 with external source.

Table 4
Material composition of the blanket.

Zones	Material and their volume fraction (%)
First wall	SS316 (50) + He (50)
Nuclear fuel	Natural uranium oxide
Coolant	Pressurized light water
Structure wall	SS316 (50) + He (50)
Tritium breeding zone	Li_2O (64) + He(36)
Reflect layer	C (80) + He (20)

Table 5
Calculation results at the first step.

	K_s		M		TBR	
Method	Value	Error (%)	Value	Error (%)	Value	Error (%)
MCNP + ORIGEN2	0.708	–	12.91	–	1.213	–
NAPTH	0.706	0.32	13.41	0.68	1.223	0.82

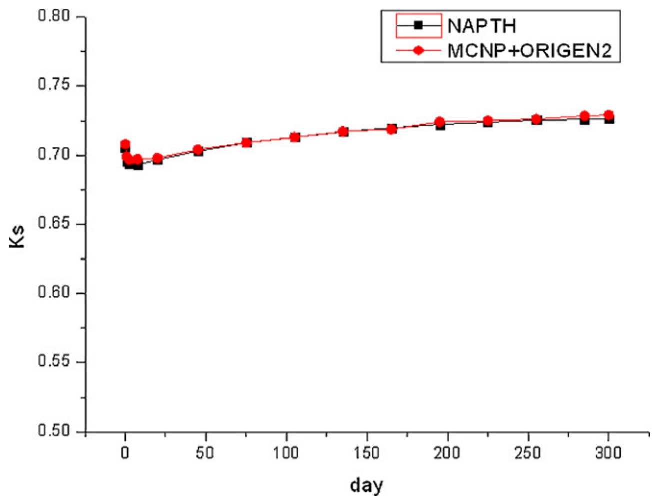

Fig. 11. K_s evolution curves.

The parameters K_s, M and TBR were calculated by NAPTH, and the reference results were from the code system MCNP + ORIGEN2 which calculated the parameters according to the equations as discussed in Section 2.4. The results of K_s, M and TBR at the first step are given in Table 5. At the first step, the relative errors of K_s, M and TBR between the two code systems are 0.32%, 0.68% and 0.82%, respectively.

The changes of K_s with time are presented in Fig. 11. At the beginning (about 2 days), K_s drops quickly owing to the absorption of neutron by the poison isotopes, and then K_s increases due to the conversion of ^{238}U–^{239}Pu. Due to the reactivity swing in the burnup history of fissionable fuels, the M and TBR change with the varying K_s. The evolution of M and TBR as a function of time are shown in Figs. 12 and 13. The trend of changes in the M and TBR are the same as K_s.

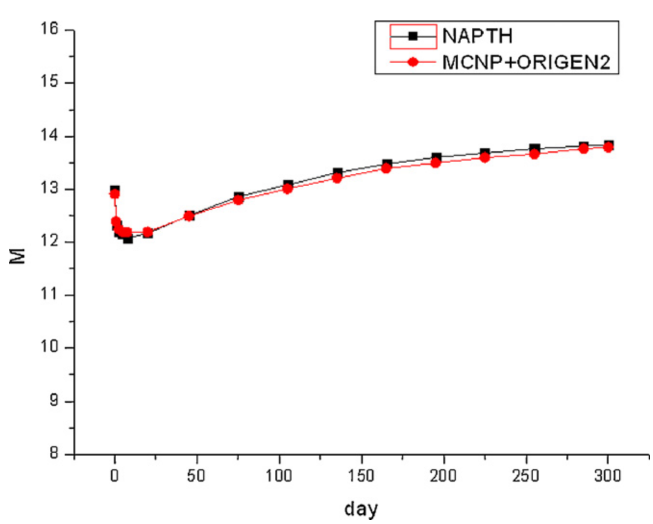

Fig. 12. Energy multiplication factor curves.

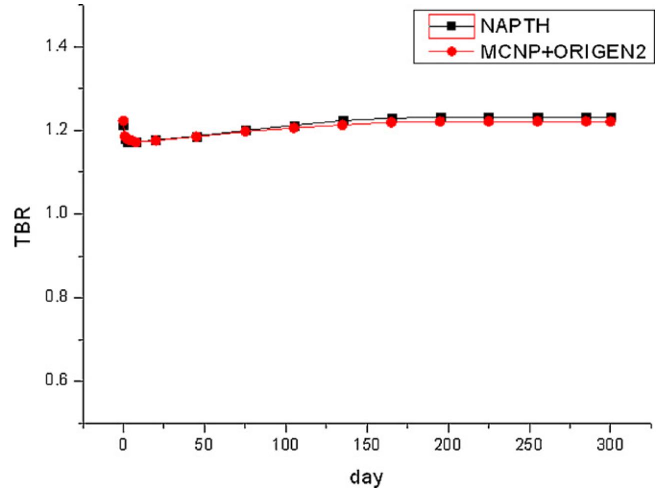

Fig. 13. TBR evolution curves.

Comparison of K_s, M and TBR calculated by NAPTH to the continuous energy MC results shows that the maximum relative errors during the lifetime are 0.60%, 0.76% and 1.34%. From the results of K_s, M and TBR it can be seen that the code NAPTH have similar precision to continuous energy calculation.

There are some factors affecting the accuracy of the results, such as assembly homogenization method and energy condensation method. In Refs. [25–27], some advanced methods were proposed to obtain accurate few-group homogeneous constants. These methods can be introduced into the code NAPTH for higher precision, in the future. Moreover, the two code systems employ different burnup calculation methods, and this factor has some effect on the results during the lifetime.

At last, the computation efficiency of the code NAPTH is discussed. In the neutronics calculations of pressure tube type FFHR, 1,000,000 particles were put both in the multigroup MC calculations and continuous energy MC calculations. 8 CPUs of 2.4 GHz were used for parallel computation. The computing time spent by NAPTH is 1.28 h including 0.83 h for lattice calculations and 0.45 h for the blanket calculation, while the time spent by MCNP + ORIGEN2 is 82.5 h. NAPTH has much better computation efficiency.

5. Conclusions

For the neutronics calculations of the hybrid reactor, the calculation methods are established and a calculation code system NAPTH is developed based on the two-step calculation scheme in this paper.

The code is tested with a PWR assembly problem and the IAEA ADS benchmark. In the former calculation, the relative errors of K_{eff} compared with WIMS9, DRAGON4 and HELIOS are 120 pcm, 30 pcm and 20 pcm at BOL, and they are 1200 pcm, 90 pcm and 230 pcm at EOL. In the latter calculation, the numerical results show good agreement with those of other participants. It indicates that the results are sensitive to the differences among the calculation tools using different libraries and calculation methods.

The code NAPTH is employed for the neutronics of a pressure tube type fusion–fission hybrid reactor with precise geometry. Some important parameters: neutron multiplication factor, energy multiplication factor and tritium breeding ratio are evaluated. The results are compared with those obtained from MCNP + ORIGEN2. The results show that NAPTH is reliable for the conceptual design of the pressure tube type hybrid reactor. The code NAPTH has much better computation efficiency than continuous energy MC code.

In the future, some work such as introducing more accurate assembly homogenization method and energy condensation method into the code is needed to get higher precision.

Acknowledgements

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