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Development of core fuel management code system for WWER-type reactors

XIA Bang-Yang [1,3,*] WANG Tao[2] XIE Zhong-Sheng[3]

([1]*National Key Laboratory for Reactor System Design Technology, Chengdu 610041, China;*

[2] *Shanghai Jiaotong University, Shanghai 200030, China;*

[3] *Xi'an Jiaotong University, Xi'an 710049, China*)

Abstract In this article, a core fuel management program for hexagonal pressurized water type WWER reactors (CFMHEX) has been developed, which is based on advanced three-dimensional nodal method and integrated with thermal hydraulic code to realize the coupling of neutronics and thermal-hydraulics. In CFMHEX, all these feedback effects such as burnup, power distribution, moderator density, and control rod insertion are considered. The verification and validation of the code system have been examined through the IAEA WWER-1000-type Kalinin NPP benchmark problem. The numerical results are in good agreement with measurements and are close to those of other international institutes.

Key words Fuel management, WWER, Nodal method

CLC number T329.2

1 Introduction

Over the past decades, although many in-core fuel management code systems for PWRs with square fuel assemblies have been developed, there are only a few codes for the cores with hexagonal assemblies (such as Russian pressurized water type WWER reactors). The Tianwan Nuclear Power Station in Jiangsu Province, China, is imported from Russia, which adopts the WWER-1000 reactor, and will be put into operation; therefore, the research of core fuel management for WWER-type reactors is very significant.

In the present study, a core fuel management program (CFMHEX) has been developed for modeling the WWER-type reactors. In this code, the in-core neutron diffusion calculation is performed by the three-dimensional multigroup hexagonal nodal code FEMHEX[1,2]. Referring to SIMULATE-3[3], a simple single channel thermal hydraulic code has been developed and integrated with FEMHEX to realize the coupling of neutronics and thermal-hydraulics. In addition, the few group cross-sections that are used in CFMHEX are generated by the hexagonal assembly code TPFAP-H[4], which is based on the transmission probability. To effectively treat the effects of feedbacks of local parameters, a multi-parameterized cross-sections method is developed and incorporated into the code CFMHEX. The major modules of CFMHEX are shown in Fig.1. It has the following major features: (1) advanced 3-D hexagonal nodal method for diffusion calculations; (2) coupled neutronics and thermal-hydraulic calculation; (3) pin power reconstruction; (4) explicit treatment of effects of multiple feedbacks of various variables on partial cross-section; (5) assembly nuclear parameters are generated by transport codes. By integrating these codes, the TPFAP-H/CFMHEX software package

*E-mail: xiabangyang@163.com

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has been developed by the authors of this study.

The verification and validation of the proposed model and code system are tested through the WWER-1000 benchmark problem issued by IAEA Coordinated Research Program (CRP)[5]. The numerical results are in good agreement with the measurements and with those of other international institutes. Deviations on power distribution and critical boron concentration are within the permitted limits of engineering requirements.

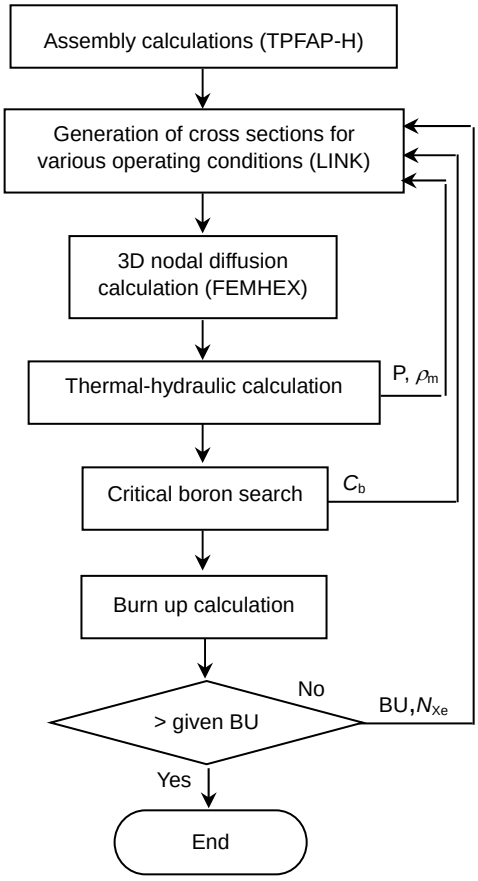

Fig. 1 Major modules of CFMHEX.

2 Diffusion calculation code FEMHEX

In FEMHEX, the intranodal flux distributions are expanded into nonseparable analytic basis functions to solve the multigroup neutron diffusion equations in hexagonal-*z* geometry. Nodes are simultaneously coupled with both the zero- and first-order partial current moments.

Considering a homogeneous hexagonal node (see Fig. 2), the matrix form of the standard 3D multigroup diffusion equations can be written as:

$$-\nabla^2\boldsymbol{\Phi}(\boldsymbol{r})+\sum(k_{\text{eff}})\boldsymbol{\Phi}(\boldsymbol{r})=0 \tag{1}$$

where $\boldsymbol{\Phi}(\boldsymbol{r})$ is the neutron flux vector, and the matrix $\sum(k_{\text{eff}})$ is a square matrix of order G with its elements $\sum_{gg'}$ as follows:

$$\sum_{gg'}=(\delta_{gg'}\sum_{tg}-\sum^{s}_{gg'}-\frac{\chi_g}{k_{\text{eff}}}\nu\sum_{fg'})/D_g \tag{2}$$

where $\delta_{gg'}$ is the Kronecker symbol, and the other notations are standard.

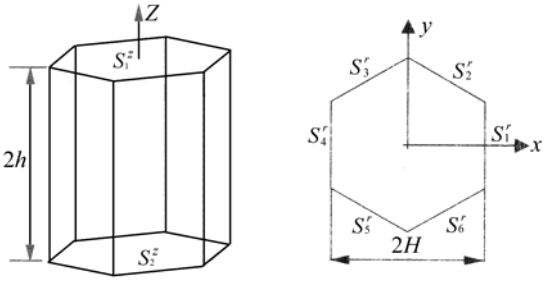

Fig. 2 Coordinate system for the hexagonal node.

The analytic solution of Eq. (1) is dictated by the eigenvalues λ_m and corresponding eigenvectors $\boldsymbol{u}_m$ of the matrix $\sum(k_{\text{eff}})$. For simplicity, only the case in which eigenvalues are real has been considered in this study. First, the following transformation is used

$$\boldsymbol{U}=[\boldsymbol{u}_1,\boldsymbol{u}_2,...,\boldsymbol{u}_G],\quad \boldsymbol{\Psi}(\boldsymbol{r})=\boldsymbol{U}^{-1}\boldsymbol{\Phi}(\boldsymbol{r}) \tag{3}$$

to reduce Eq. (1) to a decoupled form

$$-\nabla^2\Psi_m(\boldsymbol{r})+\lambda_m\Psi_m(\boldsymbol{r})=0,\quad m=1,...,G \tag{4}$$

The approximate analytic solution to Eq. (4) can be easily obtained and written as follows[1,2]:

$$\Psi_m(\boldsymbol{r})=\sum_{l=1}^{7}A_{ml}\mathrm{SN}(k_m\boldsymbol{e}_l\boldsymbol{r})+\sum_{l=1}^{7}B_{ml}\mathrm{CS}(k_m\boldsymbol{e}_l\boldsymbol{r}) \tag{5}$$

where
$$k_m=\sqrt{|\lambda_m|}, \tag{6}$$

$$\mathrm{SN}=\begin{cases}\sinh, & \lambda_m>0\\ \sin, & \lambda_m<0\end{cases}$$
$$\mathrm{CS}=\begin{cases}\cosh, & \lambda_m>0\\ \cos, & \lambda_m<0\end{cases} \tag{7}$$

$$\boldsymbol{e}_l=\cos\theta_l\,\boldsymbol{e}_x+\sin\theta_l\,\boldsymbol{e}_y,$$
$$\theta_l=(2l-3)\pi/12,\quad l=1,2,...,6 \tag{8}$$

$$\boldsymbol{e}_l = \boldsymbol{e}_z, \quad l = 7. \tag{9}$$

Then, using Eqs. (3) and (5), the intranodal flux distribution $\Phi_g(\boldsymbol{r})$ can be obtained:

$$\Phi_g(\boldsymbol{r}) = \sum_{m=1}^{G} u_{gm} \left\{ \sum_{l=1}^{7} A_{ml} \mathrm{SN}(k_m \boldsymbol{e}_l \boldsymbol{r}) + \sum_{l=1}^{7} B_{ml} \mathrm{CS}(k_m \boldsymbol{e}_l \boldsymbol{r}) \right\} \tag{10}$$

where u_{gm} is the element of the matrix $\boldsymbol{U}$ defined in Eq. (3).

The intranodal flux expansion coefficients A_{ml} and B_{ml} of Eq. (10) are determined by the nodal boundary conditions. In this method, fourteen boundary conditions are considered, which include eight surface-averaged partial currents (zero-order partial current moments) and six radial first-order partial current moments. The definitions of these interface conditions are:

$$\overline{J}_{g,k}^{r0,\pm} = \frac{1}{4}\overline{\Phi}_{g,k}^{r0} \pm \frac{1}{2}\overline{J}_{g,k}^{r0}, \quad k = 1,2,...,6 \tag{11}$$

$$\overline{J}_{g,k}^{z0,\pm} = \frac{1}{4}\overline{\Phi}_{g,k}^{z0} \pm \frac{1}{2}\overline{J}_{g,k}^{z0}, \quad k = 1,2 \tag{12}$$

$$\overline{J}_{g,k}^{r1,\pm} = \frac{1}{S_k^r}\int_{S_k^r} \mathrm{sgn}(\boldsymbol{r}) J_{g,k}^{r0,\pm}(\boldsymbol{r})\mathrm{d}\boldsymbol{r}, \quad \boldsymbol{r} \in S_k^r \tag{13}$$

where $\overline{J}_{g,k}^{r0,\pm}$ and $\overline{J}_{g,k}^{z0,\pm}$ represent radial and axial surface-averaged partial current, respectively; $\overline{J}_{g,k}^{r1,\pm}$ represents the first-order moments of the radial partial currents; and the superscript (+) and (-) represent outgoing and incoming, respectively. The values $\overline{\Phi}_{g,k}^{r0}$, and $\overline{J}_{g,k}^{r0}$ are the radial surface-averaged flux and current, respectively; whereas the values $\overline{\Phi}_{g,k}^{z0}$ and $\overline{J}_{g,k}^{z0}$ are the axial surface-averaged flux and current, respectively. In Eq. (13), the sign function $\mathrm{sgn}(\boldsymbol{r})$ is defined as:

$$\mathrm{sgn}(\boldsymbol{r}) = \begin{cases} +1 & \boldsymbol{r} \in S_k^{r,+} \\ -1 & \boldsymbol{r} \in S_k^{r,-} \end{cases} \tag{14}$$

where $S_k^{r,+}$ and $S_k^{r,-}$ are the positive and negative half-nodal surfaces, respectively.

Based on the intranodal flux expansion (10) and the definitions of the partial currents moments (11), (12), and (13), we can obtain the following two matrix equations that relate both the partial current moments and the intranodal flux expansion coefficients:

$$\overline{\boldsymbol{J}}^{-} = \boldsymbol{Q}^{-}\boldsymbol{C} \tag{15}$$

$$\overline{\boldsymbol{J}}^{+} = \boldsymbol{Q}^{+}\boldsymbol{C} \tag{16}$$

where $\overline{\boldsymbol{J}}^{-}$ and $\overline{\boldsymbol{J}}^{+}$ represent the incoming and outgoing partial current moment vector, respectively. The matrices $\boldsymbol{Q}^{+}, \boldsymbol{Q}^{-}$ are constant matrices whose elements depend on k_{eff} and the group constants of the node. The vector $\boldsymbol{C}$ is the flux expansion coefficient vector. Using Eq. (15), we have

$$\boldsymbol{C} = \boldsymbol{Q}^{-I}\overline{\boldsymbol{J}}^{-} \tag{17}$$

Then substituting Eq. (17) into Eq. (16) yields the response matrix equation as follows:

$$\overline{\boldsymbol{J}}^{+} = \boldsymbol{Q}^{+}\boldsymbol{Q}^{-I}\overline{\boldsymbol{J}}^{-} \tag{18}$$

According to the continuity conditions of $\overline{\boldsymbol{J}}^{+}$ and $\overline{\boldsymbol{J}}^{-}$ at each surface of hexagonal node, Eq. (18) forms a closed and complete system of equations. Concerning the core boundary conditions, in general, the albedos α are given for each group at the core external surfaces; so the following expression can be obtained:

$$\overline{J}_{g,k}^{s,-} = R\overline{J}_{g,k}^{s,+} \tag{19}$$

where $R = (1+2\alpha)/(1-2\alpha)$ and the superscript (s), respectively, refers to the radial and axial partial currents moments defined above. Eqs. (18) and (19) together with the aforementioned interface continuity conditions represent the equations for the iterations.

3 Thermal-hydraulic model

In CFMHEX, the reactor power, coolant density, and fuel temperature are closely coupled. In this article, a simple single-channel heat balance model is used for the thermal hydraulic calculation. The following assumptions are made: (1) the inlet flow and temperature distribution of the coolant are known;(2) the power produced by fuel rods within a node is fully transferred to the coolant; (3) coolant flow is parallel to channels, and cross flow is ignored; and (4) the pressure drop across the core is assumed to be negligible. From the assumptions it can be found that the coolant enthalpy distribution can be obtained from the heat balance of the enthalpy at the inlet, the heat generated within the node, and the enthalpy at the outlet of the node. The CFMHEX performs a coupled neu-

tronics-thermal-hydraulics iterations to determine the relationship between them. At any point during the iterative process, the power distribution can be considered known, and the coupled problem is to determine the coolant density and fuel temperature distribution for a fixed power distribution. The thermal-hydraulic results are then considered again for neutronics calculation. During each burnup step, iterations are performed till the power distribution converges.

4 Parameterized cross-section method

The macroscopic cross-sections required for the homogenized diffusion calculation of CFMHEX are provided by TPFAP-H[4]. During the fuel management calculation, the operation conditions are constantly changed. And because of the complicated cross-sections' behavior with the state variables, the nodal model should be capable of treating cross-sections as functions of node-wise burnup as well as instantaneous moderator density, boron concentration, fuel temperature, control rods insertion, etc.

The CFMHEX has the macro-cross-sections module consistent with SIMULATE-E [3]. The dependences of the macro-cross-sections are determined by performing TPFAP-H assembly calculations for each history point in advance. Based on the databases, the cross-sections can then be functionalized versus the state variables. It can be mathematically represented as a combination of "base" cross-sections and "deviation" terms:

$$\Sigma = \Sigma^{\text{base}} + \sum_i \Delta\Sigma_i \tag{20}$$

The base cross-section Σ^{base} is valid at "reference state condition" and the partial deviation term $\Delta\Sigma_i$ is computed by the perturbation of i kind of state variables compared with the reference condition, e.g. by instantaneously changing the coolant density (ρ_{m}), boron concentration (C_{b}), fuel temperature (T_{f}), etc., and can be expressed by a polynomial expression as:

$$\Delta\Sigma_i = f_1^i(x,y)\, f_2^i(z) \tag{21}$$

where x, y, z are the independent state variables considering various feedbacks (e.g. exposure, moderator density, power level, xenon concentration, etc.), and both $f_1^i(x,y)$ and $f_2^i(z)$ are polynomials up to second order. The expanding coefficients of these polynomials are generated by a linkage subroutine using the database provided by the assembly calculations. And then during the fuel management calculation, the macroscopic cross-sections for each node as functions of all instantaneous properties of the node can be automatically computed considering the various feedbacks.

5 Benchmark analysis

For validation of the code CFMHEX, the comprehensive calculations of Russian WWER-1000 reactor of Kalinin NPP have been performed. Ref. [5] gives the details of design parameters of assemblies and core, the operating conditions and loading patterns for each cycle, and the measured data of operation history given by Russia and calculation results provided by other international institutes.

The dependencies of the critical boron concentrations upon burnup (life time) of cycles 1 to 3 have been calculated, and the numerical results and comparisons with the experimental data and the results of KI (Kurchatov Institute) and VTT (Technical Research Center of Finland) are given in Tables 1–3. The symbols ε_1, ε_2, and ε_3 denote the deviations of numerical results from the measured data, and EXP denotes the measured data. From the data in these tables, it demonstrates that the calculation results are in good agreement with the measurements and with those of other institutes. It is seen from these tables that in some points, the scattering of boron concentration values are rather large. However, it can be assumed that the participant calculated the transient process of Sm-149 not in the same manner under reactor shut-down conditions for refueling, which is indicated in Ref. [5].

The assembly power distributions at different burnup for cycles 1 and 3 have been calculated and compared with the corresponding measured data. In general, the deviations of assembly power distributions for 1-3 fuel loads are less than 5%, except for few assemblies (it can reach <10%). Figs.3 and 4 show the comparisons of assembly power distribution deviations from the experimental data and the results of KI and VTT at t =235 FPD of cycle 1 and at t = 215 FPD of cycle 3, respectively.

Table 1 Comparison of critical boron concentration of WWER-1000 cycle 1 for WWER-1000 Kalinin NPP

Life time / FPD	C_b / mg·kg^{-1}						
	EXP	KI	ε_1	VTT	ε_2	CFMHEX	ε_3
6.4	1190	1201	11	1218	28	1235	45
11.5	1190	1119	-71	1149	-41	1134	-56
19.2	1100	1092	-8	-	-	1080	-20
27.5	1120	1095	-25	1109	-11	1069	-51
39.4	1120	1012	-108	1032	-88	1051	-69
47.3	1070	1070	0	1075	5	1053	-17
50.8	1070	969	-101	-	-	1012	-58
58.8	980	964	-16	-	-	951	-29
70.5	980	1120	140	1088	108	1035	55
84.6	940	865	-75	854	-86	876	-66
103.2	840	796	-44	776	-64	822	-18
123.5	750	707	-43	694	-56	752	-43
132.7	720	669	-51	658	-62	662	-58
142.2	610	639	29	619	9	629	19
154.8	590	632	42	612	22	625	35
165.7	580	519	-61	496	-84	534	-46
180.2	470	460	-10	424	-46	459	-11
197.8	420	562	142	524	104	490	70

Table 2 Comparison of critical boron concentration of WWER-1000 cycle 2 for WWER-1000 Kalinin NPP

Life time / FPD	C_b / mg·kg^{-1}						
	EXP	KI	ε_1	VTT	ε_2	CFMHEX	ε_3
0.0	1030	985	-45	1073	43	1100	70
10.0	1000	1031	31	1012	12	1003	3
13.6	940	1018	78	998	58	957	17
20.0	980	977	-3	960	-20	931	-49
25.0	980	1048	68	-	-	978	-2
27.0	980	942	-38	931	-49	906	-74
35.9	910	921	11	893	-17	895	-15
68.3	790	783	-7	750	-40	733	-57
82.3	730	705	-25	682	-48	712	-18
100.1	660	664	4	625	-35	636	-24
110.9	590	606	16	571	-19	604	14
121.8	540	562	22	524	-16	560	20
155.1	420	427	7	386	-34	381	-39
170.3	370	365	-5	324	-46	378	8
200.0	230	324	94	295	-35	220	-10
205.3	310	248	-62	214	-96	222	-88
214.8	210	282	72	-	-	252	42
217.5	230	190	-40	171	-59	193	-37

Table 3 Comparison of critical boron concentration of WWER-1000 cycle 3 for WWER-1000 Kalinin NPP

Life time / FPD	C_b / mg·kg^{-1}						
	EXP	KI	ε_1	VTT	ε_2	CFMHEX	ε_3
0.0	1010	941	-69	966	-44	1065	55
11.0	840	861	21	849	9	899	59
19.9	800	823	23	806	6	856	56
37.0	770	836	66	821	51	782	12
65.2	580	634	54	611	31	603	23
86.2	510	560	50	-	-	523	13
90.2	800	501	-299	589	-211	765	-35
95.2	520	531	11	488	-32	495	-25
97.2	470	529	59	479	9	485	15
117.7	380	429	49	393	13	390	10
120.4	380	427	47	381	1	397	17
147.0	260	310	50	271	11	266	6
155.8	240	274	34	235	-5	264	24
160.7	210	262	52	233	23	245	35
173.7	160	202	42	161	1	158	-2
182.8	120	163	43	124	4	110	-10
188.6	90	140	50	102	12	91	1
197.5	70	162	92	131	61	106	36
200.0	70	109	39	67	-3	102	32

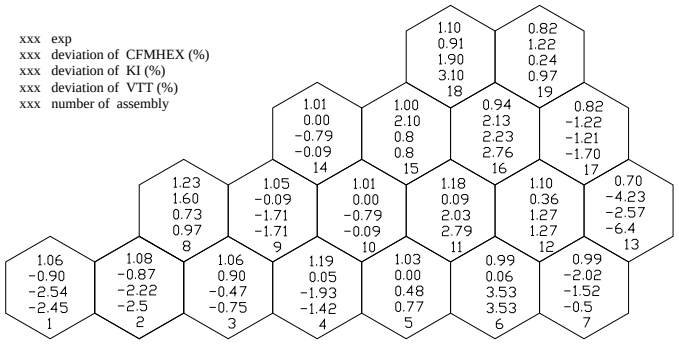

Fig. 3 Comparison of assembly power distribution (cycle 1, *t*= 235 FPD) for WWER-1000 Kalinin NPP.

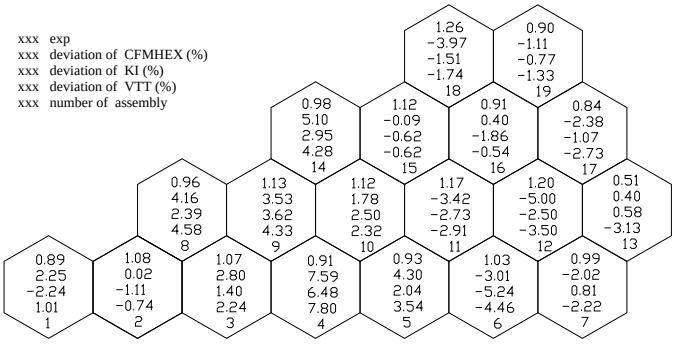

Fig. 4 Comparison of assembly power distribution (cycle 3, *t* = 215 FPD) for WWER-1000 Kalinin NPP.

6 Conclusions

In this article, an intranodal flux expansion nodal method has been developed and coupled with the thermal hydraulic code to form an in-core fuel management code package TPFAP-H/CFMHEX for WWER-type reactors. The homogenized lattice assembly parameters are generated by TPFAP-H. Primary steps in verification and validation of the code CFMHEX have been carried out and tested by the IAEA CRP benchmark problem for WWER type reactors. The numerical results demonstrate a satisfactory quantitative agreement with the measurements and are also consistent with the results of KI, VTT, and various participants of CRP. This code package is intended to be used in future analysis of WWER-type reactors.

Nomenclature

BU burnup ($MW\cdot d\cdot kg^{-1}$)

C_b critical boron concentration ($mg\cdot kg^{-1}$)

FPD characterization of a period of reactor operation in equivalent full power days

N_{Xe} number of nuclide Xe

P power distribution

Greek letters

$\Delta\sum_i$ partial deviation term of the macroscopic cross-section by perturbation of I kind of state variable compared with the reference condition (cm^{-1})

$\sum^{base}$ base macroscopic cross-section (cm^{-1})

$\sum$ macroscopic cross-section (cm^{-1})

ε_i deviation of numerical results from the reference data

ρ_m moderator density ($kg\cdot m^{-3}$)

References

1 Xia Bangyang, Xie Zhongsheng. Journal of Xi'an Jiaotong University (in Chinese), 2005, **39**(7): 762-765

2 Xia Bangyang, Xie Zhongsheng. Nuclear Power Engineering (in Chinese), 2005, **26**(5): 425-430

3 Smith K.S. Simulate-3 methodology. Studsvik /S02A-92/02-REV, 1992

4 Zhang Yin, Xie Zhongsheng. Proceeding of the Int Conf PHYSOR, Seoul, South Korea, 2002

5 IAEA-TECDOC. In-core fuel management code package validation for WWER. Vienna: IAEA, 1995: 1011-4289