Development and validation of reactor nuclear design code CORCA-3D

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Abstract

The advanced node core code CORCA-3D is one of the independent developed codes of NPIC for the nuclear reactor core design. CORCA-3D code can calculate the few-group cross section, solve the 3D diffusion equations, consider the thermal-hydraulic feedback, reconstruct the pin-by-pin power. It has lots of functions such as changing core status calculation, critical searching, control rod value calculation, coefficient calculation and so on. The main theory and functions of CORCA-3D code are introduced and validated with a lot of reactor measured data and the SCIENCE system. Now, CORCA-3D code has been applied in ACP type reactor nuclear cores design.

1. Introduction

HuaLong No. 1 reactor is one of the main types of the third generation nuclear power plant in China. With the construction of HuaLong No. 1 nuclear power plants and the need for nuclear power export, it is particularly important to develop core software with independent intellectual property right.

According to the needs of research on HuaLong No. 1 reactor and the status of core neutron analysis mostly using diffusion method, China Nuclear Power Institute has developed CORCA-3D, a reactor core nuclear design code. It can solve neutron diffusion equations of few energy groups considering the thermal-hydraulic feedback, calculate the burnup of the important actinide nuclides, fission products, and burnable poisons, obtain the power peak factor and the nuclear enthalpy rise factor through rod reconstruction calculation. CORCA-3D has also the main function for core fuel management and safety assessment on reload, such as the variable parameter calculation, critical search, control rod value calculation, the equilibrium concentrations of iodine and antimony and coefficient calculation and so on.

This paper introduces the main physical model and verification results. The reference results are proved by the measured data from the nuclear power plant and the SCIENCE system [1]. The SCIENCE system developed by the Framatome is a nuclear code package to perform the nuclear analysis of PWR cores. It consists of a set of core physics tools based on 2D lattice calculations, APOLLO2-F [2] and 3D core calculation, SMART. The SCIENCE system has a lot of engineering application experience.

2. Theory

This chapter provides main equations and methods used in the diffusion calculation, microscopic burnup, cross section calculation and rod information reconstruction in CORCA-3D code.

CORCA-3D solves the few energy group diffusion equation for the core and provides the power distribution for thermal-hydraulic calculation. The results such as moderator density, fuel temperature are for updating the cross section and step into the next diffusion calculation. After convergence of the neutronics and thermal-hydraulic coupling, CORCA-3D calculates the power peak factor and the nuclear enthalpy rise factor through the power reconstruction. Then Boron concentration searching or burn-up calculation starts.

2.1. Diffusion theory

CORCA-3D code can calculate the diffusion equations for the core of the rectangular and hexagonal assembly, and its boundary conditions allow the albedo or fixed flux boundary. Based on the rectangular Green Function nodal method [3], the multi-geometry Green Function nodal method [4] is introduced using the conformal mapping between the rectangle and the hexagon. The conformal
mapping [5] is obtained through the use of the Schwartz–Christoffel transformation [6], and the shape of the rectangle is fixed with a specific base-to-height ratio.

Three-dimension neutron diffusion equation for rectangular node after the conformal mapping takes the follows form:

\[
-D^k_s \left( \frac{\partial^2}{\partial u^2} + \frac{\partial^2}{\partial v'^2} + \frac{\partial^2}{\partial z^2} \right) \phi^k_s (u, v', z) + \sum_{i \neq s} \gamma(g) \phi^k_i (u, v', z) = \frac{\Gamma^k}{\kappa_{eff}} \sum_{k} \sum_{l} \gamma(g) \phi^k_i (u, v', z)
\]

where superscript \( k \) is the number of node, \( u, v' \) is the three-dimensional coordinate number after the conformal mapping; \( D \) is the diffusion coefficient; \( \Gamma \) is the total removal section; \( \gamma(g) \) is the Green's function of the second type of boundary condition is introduced to obtain the flux integral equation [2], the interface current coupling equation for the adjacent \( k-1, k, k+1 \) nodes (3) and the node equilibrium equation (4). (2)(3)(4) form the basic equations for diffusion calculation.

\[
\phi^k_{gu}(u) = \frac{G^k_{gu}}{\partial^2 \phi^k_{gu}} \left( \frac{\partial^2}{\partial u^2} + \frac{\partial^2}{\partial v'^2} + \frac{\partial^2}{\partial z^2} \right) \phi^k_{gu} (u, v', z)
\]

\[
= \frac{\Gamma^k}{\kappa_{eff}} \sum_{k} \sum_{l} \gamma(g) \phi^k_i (u, v', z)
\]

(1)

\[
R^k_{gu} = \frac{G^k_{gu}}{\partial^2 \phi^k_{gu}} \left( \frac{\partial^2}{\partial u^2} + \frac{\partial^2}{\partial v'^2} + \frac{\partial^2}{\partial z^2} \right) \phi^k_{gu} (u, v', z)
\]

(2)

\[
\phi^k_{gu} : \text{the traverly averaged values of flux}
\]

\[
G^k_{gu} : \text{Green Function,}
\]

\[
R^k : \text{the side length of the hexagonal node,}
\]

\[
b^k : \text{the rectangular width after the conformal mapping,}
\]

\[
A^k : \text{the radial area of the node,}
\]

\[
b^k : \text{the height of the node,}
\]

\[
f^k_{gul}, f^k_{gus} : \text{the average interfaces current flow of the left(right) side of the S direction,}
\]

\[
\bar{\psi}^k_{gu} : \text{the node average flux.}
\]

The residual weight method is used to solve the equations above. The flux is expanded into a secondary Legendre polynomial to solve the source iteration.

They can be solved with the conventional outer and inner iteration method. The inner iteration is performed on Equation (3) to solve for the currents for the given sources. With the currents available, Equation (4) can solve the node average flux \( \bar{\psi}^k_{gu} \). And the outer iteration is performed to update the flux and the source. Equation (2) can solve the new \( \phi^k_{gu} \) using the new currents, new leakage moments and the old source moments. With the new \( \phi^k_{gu} \) available, the source is updated and step into the next iteration.

### 2.2. Depletion models

The burnup module of CORCA-3D uses the Chebyshev Rational Approximation Method(CRAM) [7] and Projected Predictor-Corrector method(PPC) [8] to resolve depletion equations.

The point depletion is approximated by a system of linear equations with constant coefficients:

\[
\frac{dn_i}{dt} = -\lambda_i n_i - \phi(t) n_i \sum_k \sigma_{i,k} + \sum_j N_j (b_j \lambda_j + \sum_k \lambda_{j,k} \sigma_{i,j,k})
\]

(9)

Where \( \phi(t) \) is the neutron flux of time \( t \); \( \lambda \) is the decay constant of nuclide \( i \); \( \sigma_{i,k} \) is the microscopic cross section of nuclide \( i \) in reaction \( k \); \( b_j \) is the branching ratio for transmutation of nuclide \( j \) to nuclide \( i \) in decay reaction; \( \lambda_{j,k} \) is the branching ratio for transmutation of nuclide \( j \) to nuclide \( i \) in reaction \( k \).

Equation (9) can be converted into the matrix form:

\[
\frac{d\bar{n}(t)}{dt} = A \bar{n}(t)
\]

(10)

Equation (10) can be formally solved by the matrix exponential method yielding the simple solution:

\[
\bar{n}(t) = e^{At} \bar{n}(0)
\]

(11)
So far, CRAM has been one of the advanced point-burnup calculation algorithms, which has higher accuracy. The formula is as follows for $x \in (-\infty, 0)$:

$$e^x = a_0 + \sum_{i=1}^{k} \frac{a_i}{x - \lambda_i} = a_0 + 2\Re \sum_{i=1}^{k/2} \frac{a_i}{x - \lambda_i}$$  \hspace{1cm} (12)

Where $a_0, a_i, \lambda_i$ is the CRAM coefficients [9], $k$ is the CRAM order.

### 2.3. Homogenous cross section calculation

The macroscopic cross section in energy group $g$ of the node in core can be represented as the summation of multiplication of nuclides density of isotopes and its microscopic cross section of the energy group, which is as follows:

$$x_g^x = \sum_i N_i \sigma_{i,x}^g$$ \hspace{1cm} (13)

The depletion module of CORCA-3D considers the actinium group isotopes (such as $^{234}\text{U}$, $^{235}\text{U}$, $^{236}\text{U}$, $^{234}\text{Pu}$, $^{237}\text{Np}$, $^{238}\text{Pu}$, $^{239}\text{Pu}$, $^{240}\text{Pu}$, $^{241}\text{Pu}$, $^{242}\text{Pu}$, $^{241}\text{Am}$, $^{242}\text{Am}$, $^{234}\text{Np}$, $^{234}\text{Am}$, $^{243}\text{Cm}$, $^{244}\text{Cm}$), the fission products (such as $^{135}\text{I}$, $^{135}\text{Xe}$, $^{147}\text{Nd}$, $^{147}\text{Pm}$, $^{148}\text{Pm}$, $^{149}\text{Pm}$, $^{140}\text{Sm}$), and burnable poison (such as $^{10}\text{B}$, $^{154}\text{Gd}$, $^{155}\text{Gd}$, $^{156}\text{Gd}$, $^{157}\text{Gd}$, $^{158}\text{Gd}$). The fission products in CORCA-3D code is less than the real ones produced. A pseudo isotope is a package of the rest of fission products and all unburnable isotopes mainly in the structure material for the requirement of engineering design.

So the homogenized few group macroscopic cross section of a node is represented as:

$$\sum_{g} = \sum_{i=1}^{n} N_i \sigma_i + \text{pis} + \Delta \Sigma_{\text{rod}}$$ \hspace{1cm} (14)

The part macroscopic cross section of $\Delta \Sigma_{\text{rod}}$ denotes the influence of the insertion of the control rod, and it is the delta value between inserted and raised.

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**Fig. 1.** User interface of core load.

**Fig. 2.** User interface of reload.
The macroscopic cross section of the pseudo isotope or the microscopic cross section of an isotope is the summation of a basic part and three variable parts. The formula can be written as:

\[ X = X_b + \Delta X_m + \Delta X_f + \Delta X_H \]  

(15)

\( X \) denotes the macroscopic cross section of the pseudo isotope or the microscopic cross section of an isotope. \( X_b \) is the basic value in the reference state. \( \Delta X_m \) is the influence considering the variation of moderator density. \( \Delta X_f \) is the influence considering the variation of fuel temperature. \( \Delta X_H \) is the influence of the different burn-up history. The form of each part is polynomial. The subsection interpolation is applied for the consideration of high precision.

According to the current and history information, such as burnup, the nuclides density of isotopes, the concentration of boron, the moderator density, the effective fuel temperature, the cross section calculation module in CORCA-3D code calculates the homogenized few group macroscopic cross section.

2.4. Thermal-hydraulic feedback

CORCA-3D uses a closed-channel thermal-hydraulic feedback model and assumes no exchange of energy and mass between the channels. In single-channel thermal feedback model the inlet-flow is inputted by user, the coolant temperature, void fraction and water density are calculated along the flow direction, and the fuel effective temperature is calculated by interpolating. The single-channel thermal feedback model divides heat transfer region into single-phase convection region, subcooled boiling region, saturated
boiling region by wall temperature, subcooled and surface heat transfer coefficient. In different heat transfer region, different empirical correlation are used to calculate the heat transfer coefficient, vapor quality and void fraction.

2.5. Rod power and burnup reconstruction

The modulation method is used to reconstruct the parameters of the pin power and burnup in CORCA-3D code [10]. The node corner flux is expanded as 32-term polynomial, whose unknowns can be solved by the node and surface average flux and surface current of the adjacent nodes. The pin flux is expanded as 13-term polynomial, considering the assembly discontinuity factor and shape factor. After solving the rod cross section, we can calculate the rod power and distribution. Then the parameters such as core power peak factor and nuclear enthalpy rise factor are calculated. The rod burnup can also obtained by the polynomial expansion method.

![Graph](image)

**Fig. 6.** Comparison of the nuclear enthalpy rise factor of ACP1000 core for C1-C5.

**Table 1**

Relative deviation of the radial power distribution of ACP1000.

<table>
<thead>
<tr>
<th></th>
<th>C1 12000 MWh/tU</th>
<th>C2 6000MWh/tU</th>
<th>C3 6000MWh/tU</th>
<th>C4 0MWd/tU</th>
<th>C5 9000MWd/tU</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
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<tr>
<td>Notes:</td>
<td>relative deviation=(C-S)/S.</td>
<td>Notes: relative deviation=(C-S)/S.</td>
<td>Notes: relative deviation=(C-S)/S.</td>
<td>Notes: relative deviation=(C-S)/S.</td>
<td>Notes: relative deviation=(C-S)/S.</td>
</tr>
<tr>
<td>C1</td>
<td>-0.82% -1.00% -0.53% -0.63% -0.21% -0.15% 0.50% 1.29%</td>
<td>0.51% 0.27% 0.74% 0.61% 0.58% -0.25% -0.45% -0.25%</td>
<td>0.08% 0.22% 0.47% -0.07% 0.03% -0.48% -0.35% -0.14%</td>
<td>0.07% 0.07% 0.07% 0.15% 0.15% 0.15% 0.15% 0.15%</td>
<td>-2.27% -0.87% 0.27% -0.45% 0.48% -0.30% 1.84% -1.40%</td>
</tr>
<tr>
<td>C2</td>
<td>-1.00% -0.63% -0.76% -0.39% -0.49% 0.04% 0.23% 1.58%</td>
<td>0.28% 0.69% 0.67% 0.85% 0.29% -0.10% -0.70% -0.35%</td>
<td>0.23% 0.37% 0.20% 0.09% -0.55% -0.59% -0.23% -0.16%</td>
<td>0.48% 0.32% 0.43% -0.48% -0.28% -0.52% 1.38% 0.02%</td>
<td>-0.99% -0.31% -1.00% -1.24% 0.02% -1.30% 1.09% -1.50%</td>
</tr>
<tr>
<td>C3</td>
<td>-0.53% -0.47% -0.41% 0.15% 0.15% 0.07% 0.03%</td>
<td>-0.07% -0.38% -0.13% -0.52% -0.13% 1.52% -0.45%</td>
<td>-0.38% -0.03% -0.02% 0.03%</td>
<td>-0.18% -1.08% -1.50%</td>
<td>-0.45% -0.23% -0.20% -0.90% 0.54% 1.88% 0.58% 0.05%</td>
</tr>
<tr>
<td>C4</td>
<td>0.04% 1.58% 2.04% 0.69% 0.73% 0.06% 1.70% -0.64%</td>
<td>2.04% 1.58% 2.04% 0.69% 0.73% 0.06% 1.70% -0.64%</td>
<td>1.75% 1.64% 2.15% 1.00% 0.30% -0.20% -0.79% -0.10%</td>
<td>2.19% 2.19% 2.20% 0.18% 0.05% -1.00% -1.43% -1.50%</td>
<td>-2.07% -0.87% 0.27% -0.45% 0.48% -0.30% 1.84% -1.40%</td>
</tr>
<tr>
<td>C5</td>
<td>-1.08% -0.53% -0.47% -0.15% 0.15% 0.07% 0.03%</td>
<td>-0.08% -0.31% -1.00% -1.34% 0.02% -1.30% 1.09% -1.50%</td>
<td>1.71% -0.85% -1.43% -1.12% -1.88%</td>
<td>-0.61% -1.08% -1.50%</td>
<td>-0.45% -0.23% -0.20% -0.90% 0.54% 1.88% 0.58% 0.05%</td>
</tr>
</tbody>
</table>
2.6. User interface

CORCA-3D uses a graphical interface to view input modeling and calculation results, and user can use the mouse for the core loading and reload modeling. The output interface can directly display the physical parameters such as the fuel consumption, power, and important nuclide nuclear density of the component, and display it with cloud distribution. CORCA-3D code has the good human computer interaction and user experience. The Figs. 1–3 show the user interface of core load, reload and output.

3. Validation

3.1. HuaLong No. 1 core

The HuaLong No.1 core, ACP1000 is the 3rd generation nuclear power plant of a pressurized light water type that uses a low concentration of uranium dioxide fuel. The core consists of 177 fuel assemblies. We calculate the first five-cycle core of the ACP1000 and compare the main results with the SCIENCE system. Among them, the few group cross section library used by CORCA-3D code is calculated by the assembly neutron analysis code KYLIN-II [11] based on the MOC method.

The core in this example is in the hot full power operation state(HFP) and all control rods are raised out of the core(ARO). We maintain the critical state by changing the boron concentration of the core. Fig. 4 shows the absolute deviations of the critical boron concentration with the burn-up in the first five-cycle of the ACP1000. Compared with SCIENCE system, the maximum absolute deviation of critical boron concentration during the full life period is less than 50 ppm. At the same time, the power peak factor, Fq and the nuclear enthalpy rise factor, Fdh are listed in Fig. 5 and Fig. 6 at the status of the balance xenon. The maximum relative deviation of Fq and Fdh are less than 5%(individual points less than 6%) compared to SCIENCE system. The results are in good agreement.

Table 1 shows the maximum relative deviations of the radial relative power of the core at some burnup times such as 0 MWd/ tU(BOL), 6000-9000 MWd/tU(MOL)and 12000 -17000MWd/tU(EOL) between CORCA-3D code and SCIENCE system. All of them, the maximum relative deviations are less than 5%.

3.2. The core consisting of 157 fuel assemblies

Unit 1 of LingAo nuclear plant and Unit 1 of FangJiaShan nuclear plant are the typical second generation plus nuclear power plants with 157 fuel assemblies in core. We choose the cycle 1 to cycle 14 of LingAo unit 1 plant and cycle 1 and cycle 4 of FangJiaShan unit 1 plant as the calculation object and calculate the critical Boron concentration, radial power distribution, Fq, Fdh, burnup distribution at end of every cycle life using CORCA-3D code. The reference data of the critical Boron concentration is the measured data provided by the nuclear power plant. The reference data about radial power distribution, Fq, Fdh and the burnup distribution at the end of life are the results of SCIENCE system. We show the statistical analysis of the comparison results.

The comparison results \( x_1, x_2, \cdots, x_n \) are the samples. \( n \) is the sample number. The average value of the sample is \( \bar{x} = \frac{1}{n} \sum_{k=1}^{n} x_k \). The mean deviation of the sample is \( \frac{1}{n} \sum_{k=1}^{n} |x_k - \bar{x}| \). The standard deviation of the sample is \( \sqrt{\frac{1}{n} \sum_{k=1}^{n} (x_k - \bar{x})^2} \). The confidence interval of 95% means 95% probability of the values belongs to this interval.
3.2.1. Critical Boron concentration

The sample number of the absolute deviation of the critical Boron concentration is 167. The average value of the sample is $-4.8$ ppm. The mean deviation of the sample is $11.4$ ppm. The standard deviation of the sample is $15.4$ ppm. The confidence interval of 95% is $[-43.0$ ppm, $27.0$ ppm]. Fig. 7 shows the statistical chart of relative deviation of critical Boron concentration.

3.2.2. Radial power distribution

The sample number of the relative deviation of the radial relative power distribution at high level (relative power $\geq 0.9$) is 29998. The average value of the sample is $-0.17\%$. The mean deviation of the sample is $0.58\%$. The standard deviation of the sample is $0.73\%$. The confidence interval of 95% is $[-1.65\%, 1.17\%]$. The sample number of the relative deviation of the radial relative power distribution at low level (relative power $< 0.9$) is 10326. The average value of the sample is $-0.18\%$. The mean deviation of the sample is $0.92\%$. The standard deviation of the sample is $1.20\%$. The confidence interval of 95% is $[-2.99\%, 1.69\%]$. Fig. 8 shows the statistical chart of relative deviation of radial relative power distribution.

3.2.3. Power peak factor

The sample number of the relative deviation for the power peak factor, $F_q$ is 274. The average value of the sample is $-0.01\%$. The mean deviation of the sample is $1.05\%$. The standard deviation of the sample is $1.28\%$. The confidence interval of 95% is $[-2.18\%, 2.71\%]$. Fig. 9 shows the statistical chart of relative deviation of the power peak factor.

3.2.4. Nuclear enthalpy rise factor, $F_{dh}$

The sample number of the relative deviation of the nuclear enthalpy rise factor, $F_{dh}$ is 274. The average value of the sample is $0.52\%$. The mean deviation of the sample is $0.70\%$. The standard deviation of the sample is $0.92\%$. The confidence interval of 95% is $[-1.34\%, 2.43\%]$. Fig. 10 shows the statistical chart of relative deviation of the nuclear enthalpy rise factor, $F_{dh}$.

3.2.5. Burnup distribution at end of life

The sample number of the absolute deviation of the assemblies burnup at end of life is 2826. The average value of the sample is 39 MWd/tU. The mean deviation of the sample is 63 MWd/tU. The standard deviation of the sample is 77 MWd/tU. The confidence interval of 95% is $[-12.3$, $35.7$]. Fig. 11 shows the statistical chart of absolute deviation of burnup.
interval of 95% is $[-126 \text{ MWd/tU},168 \text{ MWd/tU}]$. Fig. 11 shows the statistical chart of the absolute deviation of the assemblies burnup at end of life.

3.3. Benchmark for hexagonal geometry core

The diffusion module in CORCA-3D code can be used to calculate the hexagonal geometry core such as IAEA, VVER-1000, and VVER-440 [12]. The Fig. 12 shows the comparison results of CORCA-3D code for the benchmark problem. Reference solutions are calculated by the finite difference code. The absolute deviations of the eigenvalue, $K_{eff}$ are less than 50pcm, and the relative deviations of radial relative power of the average assembly are less than 2%.

4. Conclusion

This paper introduces the three-dimension core neutron analysis code CORCA-3D developed by the China Nuclear Power Research and Design Institute. We use CORCA-3D code to calculate the cycle 1 to cycle 5 of the ACP1000 core, the cycle 1 to cycle 14 of LingAo Unit 1 core, the cycle 1 to cycle 4 of FangJiaShan Unit 1 core, and some benchmark problems for hexagonal core. The comparison results between CORCA-3D code and the measured data, the reference values and SCIENCE system agree well. Now CORCA-3D code has been applied for nuclear power plants of the second and third generation in China. We are also developing the functions for the core with the hexagonal geometry arrangement. Our goal is to make the CORCA-3D code important software for the engineering design of rectangle and hexagonal nuclear power plants core.

Appendix A. Supplementary data

Supplementary data to this article can be found online at https://doi.org/10.1016/j.net.2019.05.015.

References