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Validation of PWR Neutronics Code Package TORCH V2.0 With Nuclear Power Plant Measurements

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This article presents the verification and validation (V and V) of PWR neutronics code package TORCH V2.0 with nuclear power plant (NPP) measurements. The advanced nuclear power engineering design software, TORCH V2.0, was developed by the Nuclear Power Institute of China (NPIC), China National Nuclear Corporation (CNNC). Based on the two-step calculation scheme, TORCH V2.0 mainly contains lattice physics code for assembly homogenization, link calculation code for few-group constant parameterization, and core simulation code for few-group core calculation. The calculation modules of each code were already verified against various benchmark problems, whereas this article focuses on the V and V of linked code system. The measured values of the reactor startup physics test and NPP operation from six PWR NPPs (Daya Bay NPP, Ling Ao NPP, Fangjiashan NPP, Qinshan NPP, Hainan Changjiang NPP, and Fuqing NPP) were utilized to perform the comparison and analysis of V and V. Compared parameters of the reactor startup physics test include critical boron concentration, control rod integral value, boron differential value, and isothermal temperature coefficient. Compared parameters of the NPP operation contain critical boron concentration, assembly-wise power distribution, hot spot factor, and nuclear enthalpy rise factor. The results show that the software TORCH V2.0 has reliable calculation ability and can be applied in the PWR nuclear power engineering design which is based on square fuel assembly.

Keywords: validation and verification, TORCH V2.0, nuclear power plant operation data, measurements, startup physics test

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INTRODUCTION

A conventional two-step approach of a transport calculation and a nodal diffusion calculation, such as CASMO/SIMULATE, PARAGON (or PHOENIX)/ANC, and APOLLO/SMART (Liu and Meliksetian, 1986; Studsvik of American, 1994; Studsvik of American, 1995; Westinghouse Electric Company, 2005; Adrien, 2014; Vidal et al., 2014; Zhang et al., 2018), has been used in the light water reactor core design for decades. These conventional code systems have been in use for commercial PWR core designs for a long time, and they have been upgraded continuously based on numerous core calculations. However, the conventional code systems usually adopt methodologies developed in the past. A PWR core design software package named TORCH V2.0 has been developed by the Nuclear Power Institute of China (NPIC), CNNC. Many advanced methodologies (Zhang et al., 2019) have

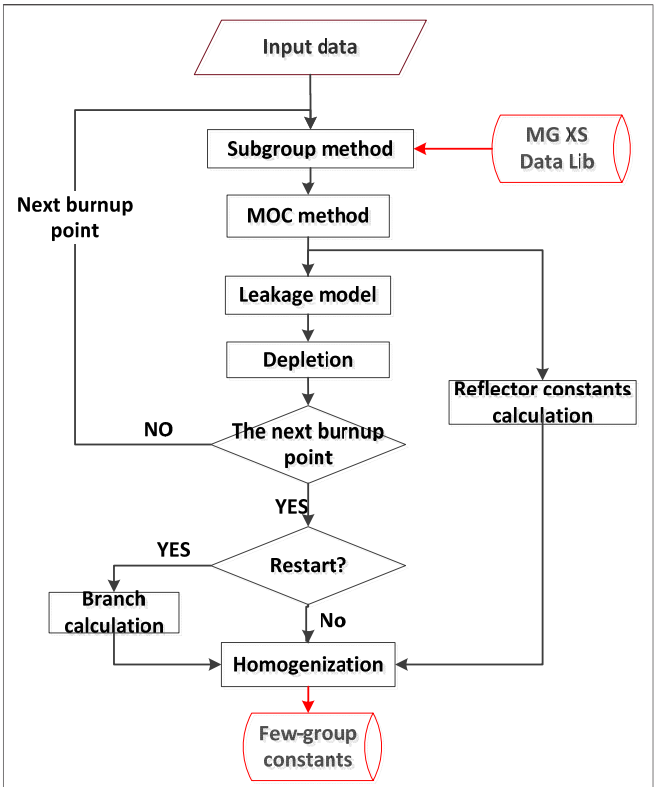

FIGURE 1 | Main calculation flow chart of the KYLIN V2.0 program.

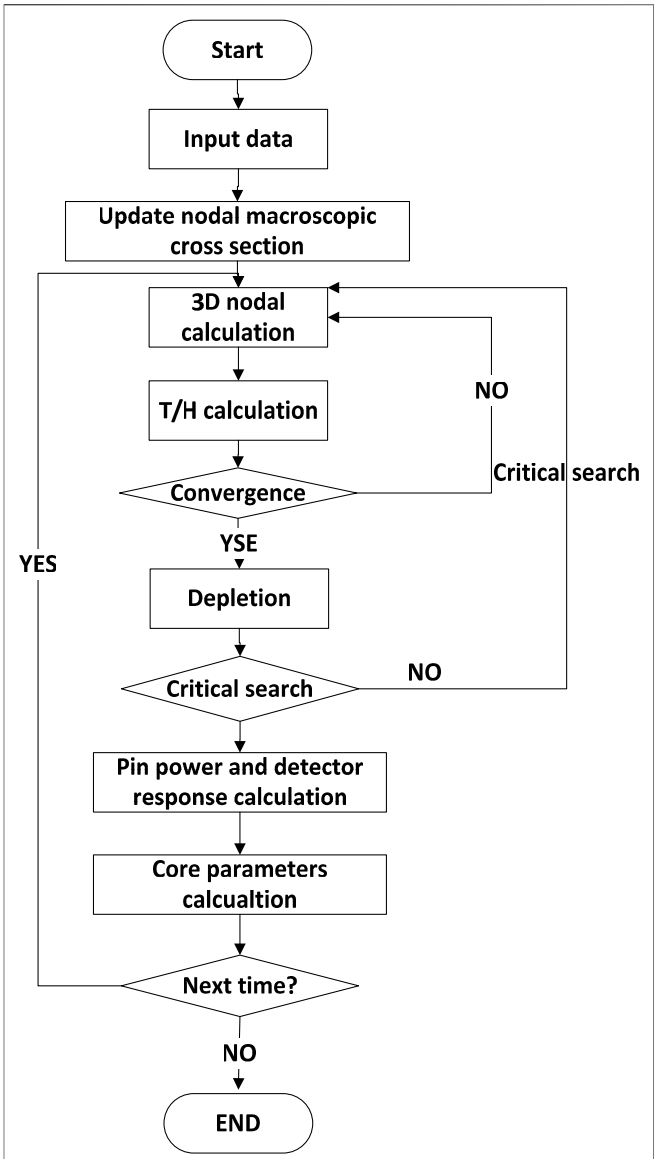

FIGURE 2 | Main calculation flow chart of the CORCA-3D code.

been implemented in TORCH V2.0 to enhance the accuracy and performance. TORCH V2.0 has been developed to be a multi-scale, multi-physics analysis code system; thus, it can be a platform in terms of a neutronics code for coupling with thermal/hydraulic code and fuel performance code. TORCH V2.0 is a property code which is mainly used for the PWR nuclear power engineering design with square fuel assembly, and it is developed along with China's third-generation nuclear power HPR1000.

The results show that the software TORCH V2.0 has reliable calculation ability and can be applied in the PWR nuclear power engineering design which is based on square fuel assembly.

TORCH V2.0 is a PWR core design software package, which mainly includes a lattice physics code named KYLIN V2.0 (Tu et al., 2016; Chai et al., 2017), linking a calculation code named PACFAC and a few-group core simulation code named CORCA-3D (An et al., 2019). KYLIN V2.0 has powerful geometric processing ability, which can perform one-step two-dimensional neutron transport calculation and analysis on various fuel assemblies with complex structures in advanced nuclear reactors, and is mainly used to provide the few-group assembly-homogenized parameters for the three-dimensional core computing software CORCA-3D. PACFAC is a linking code which uses the interpolation/fitting method to parameterize the few-group assembly-homogenized parameters generated by KYLIN V2.0 and provides the polynomial parameters to CORCA-3D to calculate the few-group constants which can be determined by the core state. CORCA-3D software adopts an advanced nodal method to solve the few-group diffusion equation and can perform core depletion calculation, power reconstruction, and physical and thermal coupling calculations.

The reactor cores of six PWR nuclear power plants (Daya Bay NPP, Ling Ao NPP, Fangjiashan NPP, Qinshan NPP, Hainan Changjiang NPP, and Fuqing NPP) have been 85 cycles for 14 NPP units reactor startup physics tests and a total of were simulated by TORCH V2.0. Compared parameters of the reactor startup physics test include critical boron concentration, control rod integral value, boron differential value, and isothermal temperature coefficient. Compared parameters of the nuclear power plant (NPP) operation include critical boron concentration, assembly-wise power distribution, hot spot factor, and nuclear enthalpy rise factor.

TABLE 1 | The details information of simulated six PWR NPPs.

NPP	Unit	Simulated cycles of startup physics test	Simulated cycles of reactor operation
Daya Bay NPP	Unit 1	9–20	9, 12, 15, 17
	Unit 2	9–20	9, 12, 15, 17
Ling Ao NPP	Unit 1	5–16	—
	Unit 2	5–15	
Fangjiashan NPP	Unit 1	1–4	1–4
	Unit 2	1–4	1–4
Qinshan NPP	Unit 1	1–15	1–15
	Unit 3	1–8	1–8
Hainan Changjiang NPP	Unit 1	1–4	1–4
	Unit 2	1–3	1–3
Fuqing NPP	Unit 1	1–4	1–2
	Unit 2	1–3	1–2
	Unit 3	1–3	1
	Unit 4	1–2	—

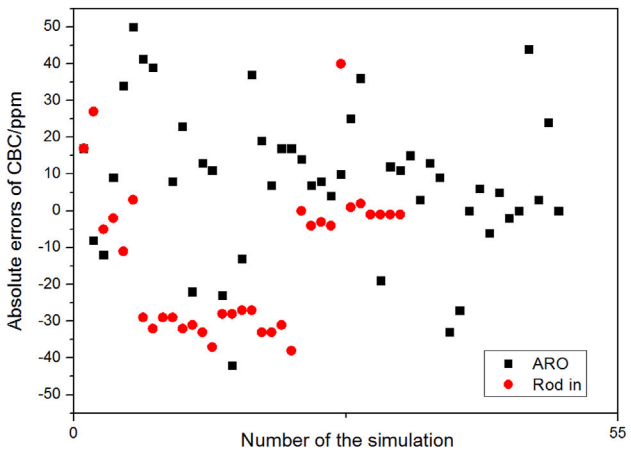

FIGURE 3 | Results of critical boron concentration.

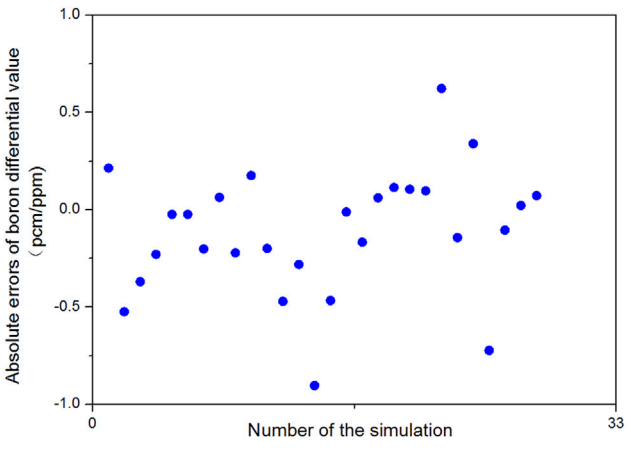

FIGURE 5 | Results of the boron differential value.

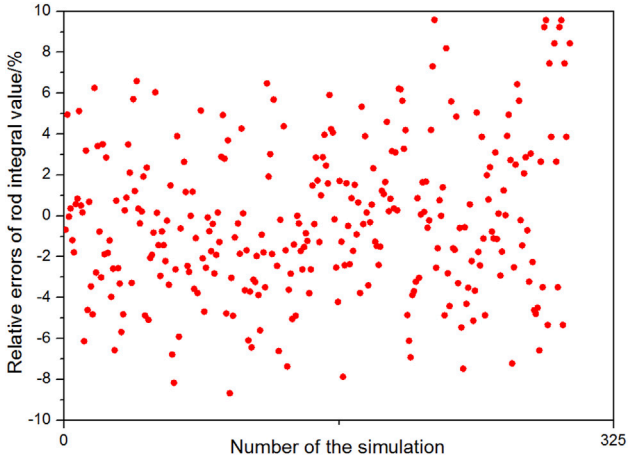

FIGURE 4 | Results of the control rod integral value.

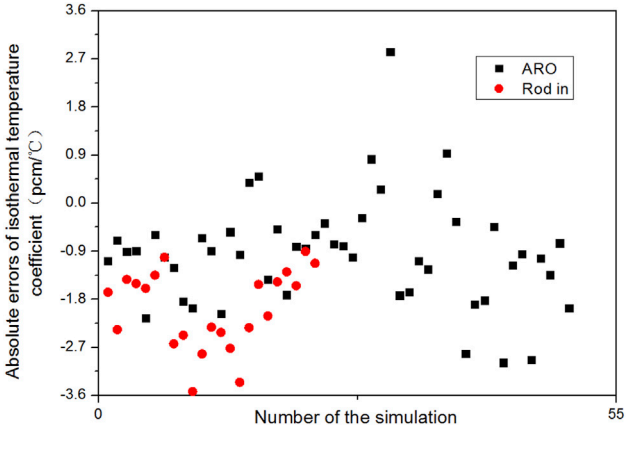

FIGURE 6 | Results of the isothermal temperature coefficient.

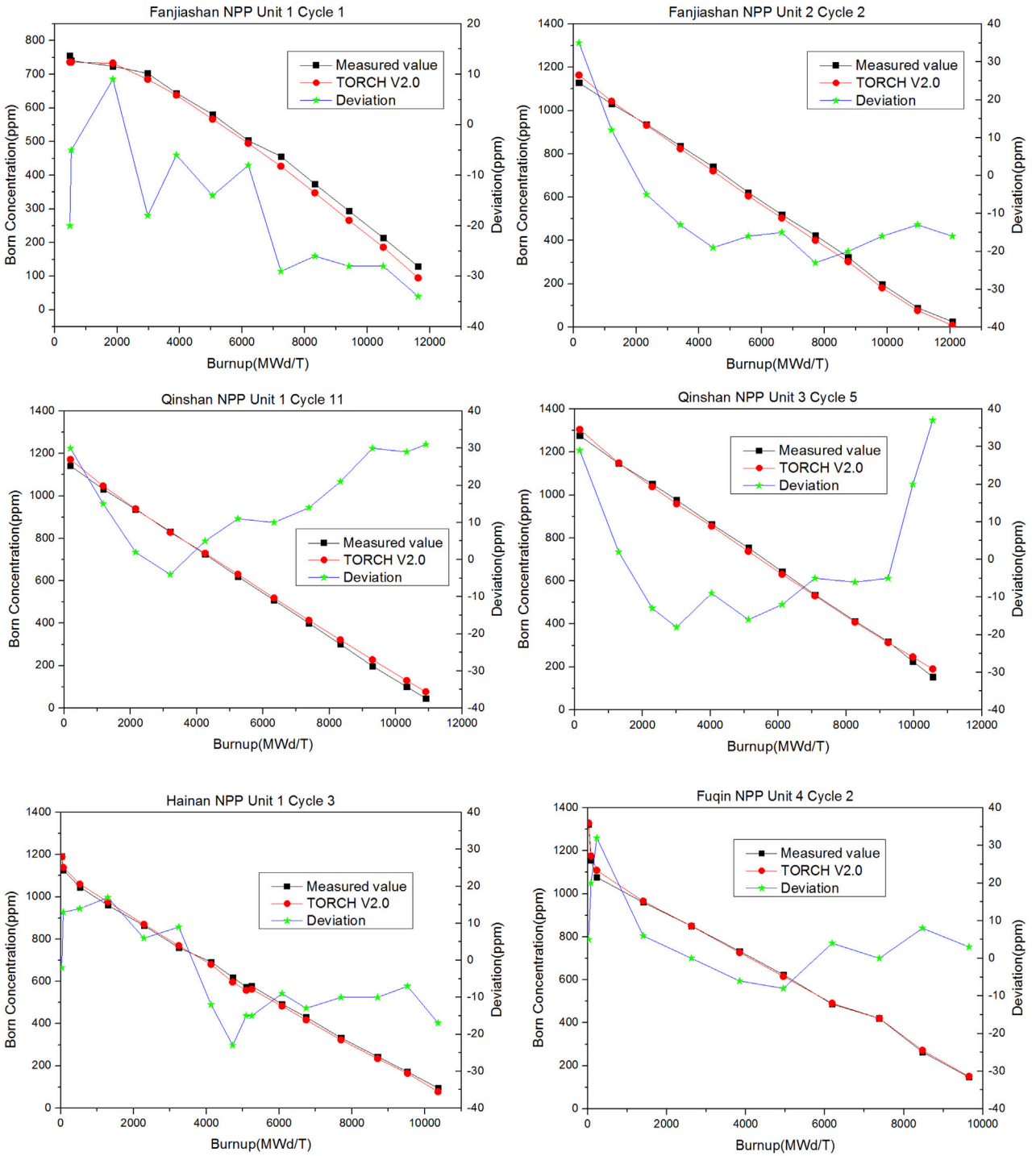

FIGURE 7 | Partial results of critical boron concentration.

CHARACTERISTICS OF TORCH V2.0

TORCH V2.0 adopts a two-step approach to simulate PWR cores. The one-step direct heterogenous calculation code KYLIN V2.0 based on a fine-group structure is used. KYLIN V2.0 is used to generate homogenized parameters, including homogenized cross-section, diffusion coefficients, and discontinuity factors. Considering the fact that a certain state required by the reactor core calculation may be different from the ones provided by KYLIN V2.0, a process is required to provide a table between those neutronics few-group constants and state parameters based on the discrete points provided by lattice

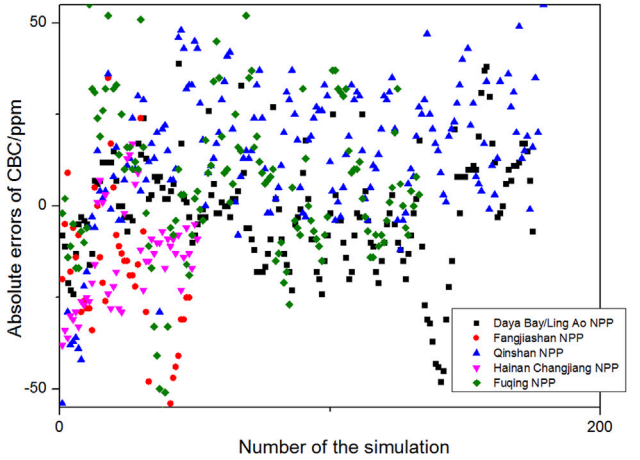

FIGURE 8 | Results of critical boron concentration.

calculations. The code named PACFAC is used to provide the table for the core simulator CORCA-3D. The neutron diffusion equation would be calculated for core fuel management in CORCA-3D.

KYLIN V2.0 Code

The flow chart of the lattice physics code KYLIN V2.0 is shown in **Figure 1**; the program is mainly used for neutron simulation calculation and analysis of two-dimensional problems and provides assembly-homogenized parameters for three-dimensional core simulation software. The KYLIN V2.0 program can select the 45-/190-group structure of multi-group cross-section libraries for calculation. It adopts an advanced subgroup method (Nikilaev et al., 1970; Hebert, 2004) to solve the effective resonance cross-section of resonance nuclides, adopts an MOC method (Hong and Cho, 1998; Kosaka and Takeda, 2004) to calculate the neutron transport of complex structure geometry, and adopts a generalized coarse mesh finite difference (GCMFD) acceleration method to accelerate

TABLE 2 | Results of Daya Bay NPP (Unit 1 Cycle 11) assembly-wise power.

Burnup (MWd/tU)	Relative assembly power ≥0.9		Relative assembly power<0.9	
	Max relative error (%)	Acceptance standard (%)	Max relative error (%)	Acceptance standard (%)
250	2.71	±5	−2.20	±8
1,492	2.56	±5	1.85	±8
2,610	3.27	±5	2.91	±8
3,810	3.48	±5	3.10	±8
5,000	3.63	±5	3.29	±8
6,250	4.19	±5	3.90	±8
7,494	3.68	±5	3.34	±8
8,650	4.08	±5	4.14	±8
11,090	3.62	±5	3.74	±8
12,289	3.02	±5	3.17	±8
13,529	2.79	±5	2.49	±8
14,456	3.19	±5	−2.42	±8
15,929	3.09	±5	−2.45	±8
17,128	3.11	±5	−2.48	±8
18,746	3.15	±5	−2.56	±8

TABLE 3 | Results of Ling Ao NPP (Unit 2 Cycle 7) assembly-wise power.

Burnup (MWd/tU)	Relative assembly power ≥0.9		Relative assembly power<0.9	
	Max relative error (%)	Acceptance standard (%)	Max relative error (%)	Acceptance standard (%)
220	2.77	±5	4.07	±8
1,260	2.75	±5	3.93	±8
2,420	2.75	±5	3.69	±8
3,700	2.85	±5	3.38	±8
4,970	3.02	±5	3.43	±8
6,220	3.04	±5	3.28	±8
7,340	−2.96	±5	3.11	±8
8,620	−2.80	±5	3.04	±8
9,860	−2.66	±5	3.02	±8
11,000	2.84	±5	3.39	±8
12,240	2.73	±5	3.25	±8
12,920	−2.79	±5	3.23	±8

TABLE 4 | Results of Fangjiashan NPP (Unit 2 Cycle 4) assembly-wise power.

Burnup (MWd/tU)	Relative assembly power ≥0.9		Relative assembly power<0.9	
	Max relative error (%)	Acceptance standard (%)	Max relative error (%)	Acceptance standard (%)
179	1.84	±5	2.32	±8
1,216	1.46	±5	2.09	±8
2,332	2.19	±5	2.91	±8
3,408	2.67	±5	3.58	±8
4,451	3.05	±5	4.05	±8
5,576	3.21	±5	3.96	±8
6,642	3.69	±5	4.87	±8
7,711	3.61	±5	4.84	±8
8,755	3.63	±5	4.99	±8
9,837	3.66	±5	4.68	±8
10,966	3.77	±5	5.05	±8

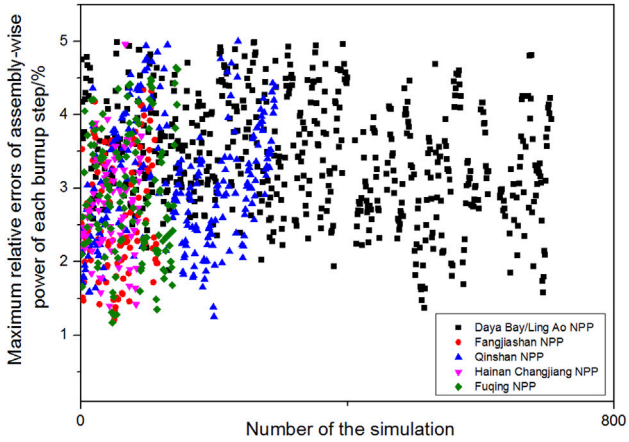

FIGURE 9 | Results of assembly-wise power (greater than 0.9).

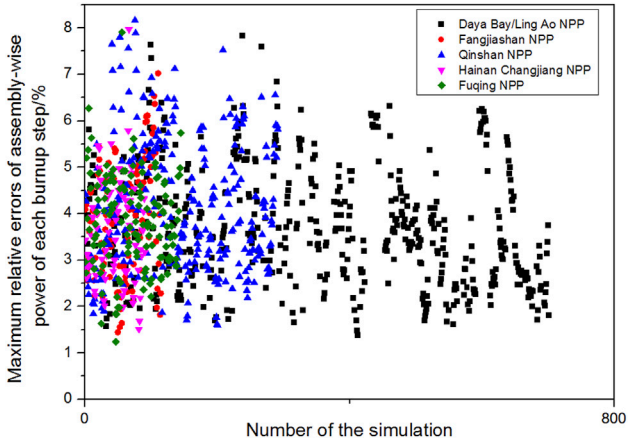

FIGURE 10 | Results of assembly-wise power (less than 0.9).

neutron transport calculation. The depletion calculation is carried out based on the improved predictive-corrected critical-fuel consumption iterative method (Yamamoto et al., 2009), and the burnup equation is solved by the Chebyshev method. At the same time, for the convenience of users, the program also has a graphical modeling tool for components with complex structure geometry and a post-processing display tool.

Specifically, KYLIN V2.0 software mainly has the following characteristics:

1) It can accurately describe geometric shape, mesh division, material distribution of various grids, or components with different geometric structures (including the bar bundle type and plate type) and provide users with clear and convenient input methods.
2) It can accurately read the input parameters necessary for grid (component) calculation, including multi-group constant library, temperature of each region, and resonance effective temperature of important nuclides.
3) It can accurately deal with complex geometric resonance problems and can also consider the influence of fuel core temperature, multi-resonance nuclide interference, and other influence factors.
4) The fine energy spectrum and homogenized cross-section parameters of various types of grids (assemblies) can be solved by using a fine energy group structure.
5) It can deal with heavy nuclear burnup chains, such as uranium series, plutonium series, and fine fission product chains, and can carry out burnup calculation of important combustible poisons (such as boron and gadolinium).
6) One-step two-dimensional heterogenous multi-assembly calculations can be performed, and restart calculations such as continuous calculation and variable working conditions can be performed.
7) It can output main calculation results, such as infinite multiplication coefficient, relative power distribution, and nuclear density of important nuclides.

PACFAC Code

The PACFAC code adopts the interpolation/fitting method to parameterize the few-group assembly-homogenized parameters generated by KYLIN V2.0 and provides the polynomial parameters to CORCA-3D to calculate the few-group constants which can be determined by the core state (Li et al., 2016). The main functions of PACFAC contain the few-group constant parameterization model of few-group constants and the inverse

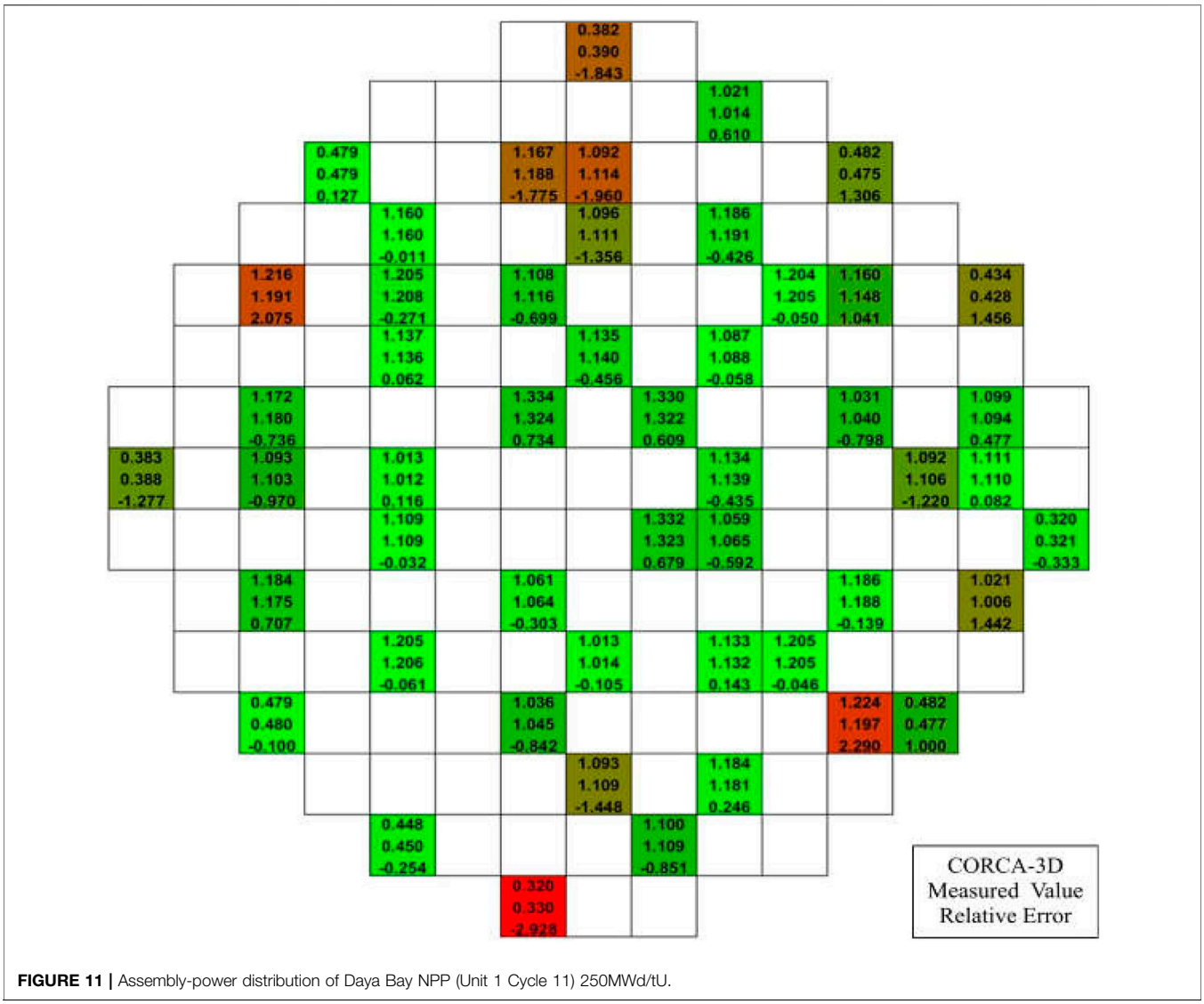

FIGURE 11 | Assembly-power distribution of Daya Bay NPP (Unit 1 Cycle 11) 250MWd/tU.

calculation model of cross-section parameters. The former provides the polynomial parameters of the interpolation/fitting relationship between the few-group constants (few-group cross-section, shape factor, and so on) and core state parameters. The inverse calculation model is mainly used to determine the few-group constants in the core simulator, which is closely related to polynomial parameters in the interpolation/fitting relation. The parameterization formulation of PACFA is shown in the following equation:

$$\sum^{actual} = \sum_{i=1}^{n} N_i^{actual} \sigma_i (DM, Bu, FT, \ldots) + \sum^{pis} (DM, \ldots) + \Delta\Sigma_{rod} (DM, \ldots).$$

Here, $\sum^{pis}$ means the contribution of pseudo-nuclides to the macroscopic cross section, which is caused by the difference between the species of nuclides in the burnup chain of CORCA-3D and burnup chain of KYLIN V2.0. $\Delta\Sigma_{rod}$ means the influence component of the control rod on the cross section.

CORCA-3D Code

The main calculation flow chart of CORCA-3D is shown in **Figure 2**, which is mainly used for the simulation of three-dimensional steady-state core, and is one of the most important calculation software for core fuel management. CORCA-3D can carry out the diffusion calculation based on the few-group constants calculated *via* the homogenized cross-section parameter library, the burnup calculation of important actinides, fission products, and combustible poisons. CORCA-3D can perform thermal-hydraulic feedback calculation, pin-power reconstruction calculation, and is able to calculate the equilibrium concentration of iodine and xenon. CORCA-3D can read multiple-cycle databases for refueling core modeling calculation. At the same time, it has the basic functions of core design, such as coefficient calculation, variable parameter calculation, boron concentration/control rod position critical search calculation, and control rod differential/integral value calculation.

The CORCA-3D code mainly has the following characteristics:

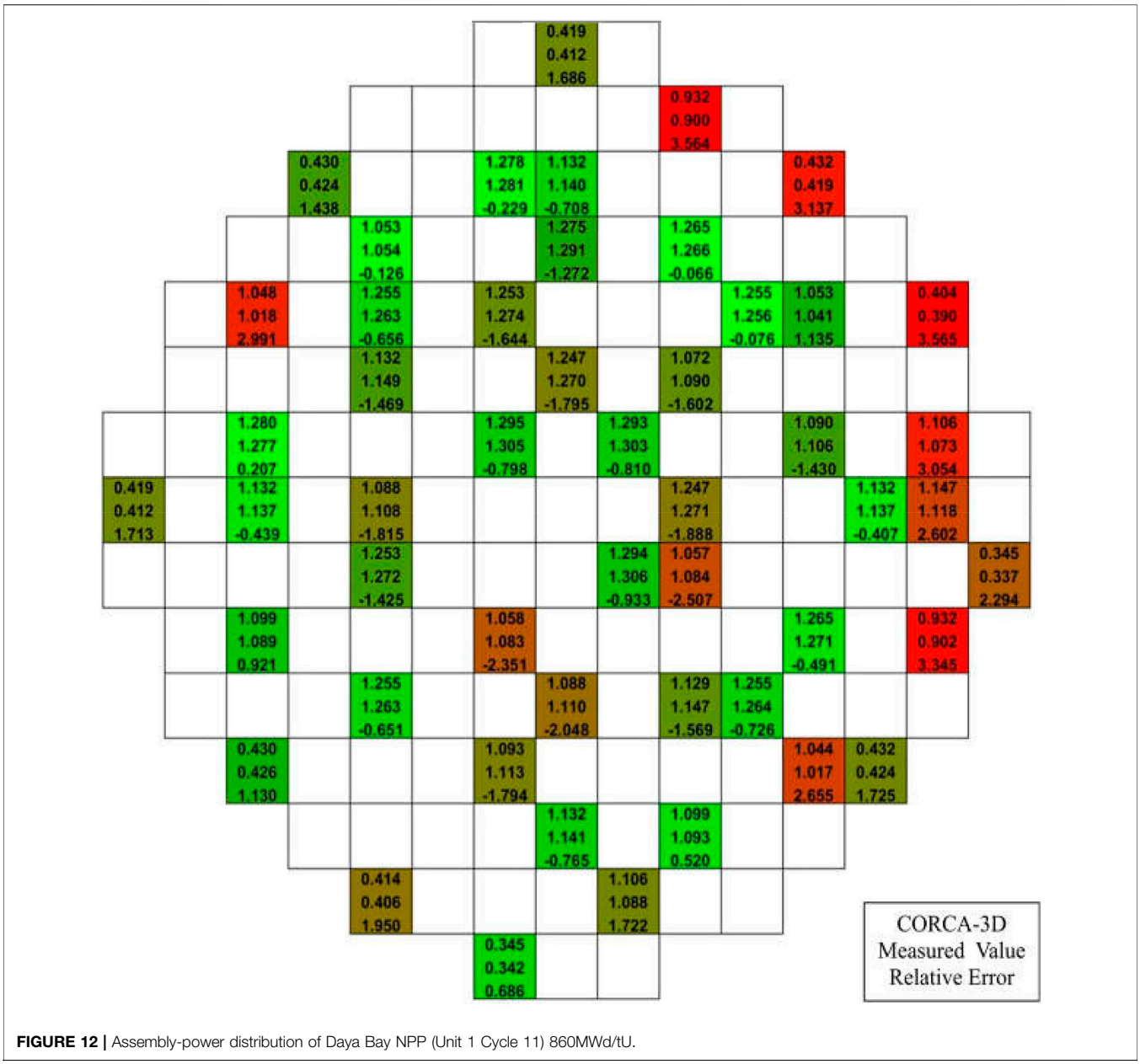

FIGURE 12 | Assembly-power distribution of Daya Bay NPP (Unit 1 Cycle 11) 860MWd/tU.

1) According to the current core information and historical information of the component (including burnup depth, boron concentration, effective fuel temperature, moderator density, xenon concentration, and control rod status), the few-group cross-section parameters of each segment in the core can be calculated from the multi-parameter cross-section library provided by the linking code PACFAC.
2) It can quickly and accurately solve the diffusion equation of few groups (two groups and four groups) in the core.
3) The power reconstruction calculation of rectangular geometry can be carried out, and the pin-power distribution and the detector response can be obtained.
4) It can solve the single-channel thermal hydraulic equation of the reactor core and calculate the enthalpy field distribution of the coolant in the reactor;
5) It can accurately solve the burnup equation of important nuclides in the reactor core and calculate the accumulation and consumption process of important nuclides (such as actinides and combustible poisons) in the reactor.
6) It can calculate the equilibrium concentration of xenon.
7) It is can search boron concentration or rod position of the control rod group to make the core eigenvalue reach the target value;
8) Parameters such as boron concentration, relative power, and control rod position can be changed at any burning time for calculation.

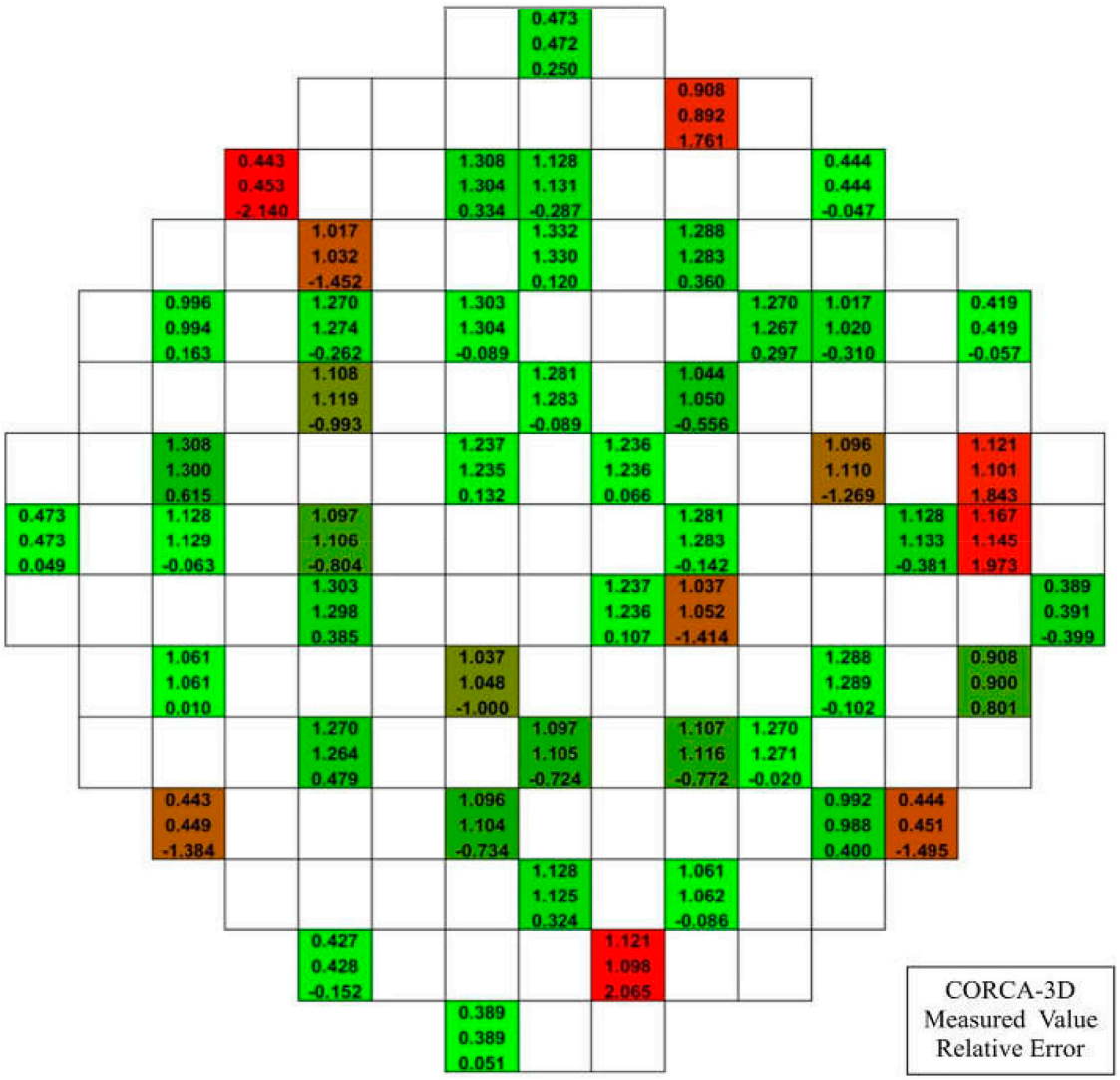

FIGURE 13 | Assembly-power distribution of Daya Bay NPP (Unit 1 Cycle 11) 18746MWd/tU.

VERIFICATION AND VALIDATION RESULTS

The reactor cores of six PWR NPPs are simulated. The analyzed NPPs shown in **Table 1** are Daya Bay NPP (DYBU1C9-WRS-410100-BG1 Rev. 0, 2010), Ling Ao NPP (LAU1C5-WRS-410100-BG1 Rev. 0, 2009), Fangjiashan NPP (FJSU1C1-WRS-410100-BG1 Rev. 0, 2016), Qinshan NPP (QSU1C1-WRS-410100-BG1 Rev. 0, 2005), Hainan Changjiang NPP (CJU1C1-WRS-410100-BG1 Rev. 0, 2016), and Fuqing NPP (FQU1C1-WRS-410100-BG1 Rev. 0, 2016). Part of reactor startup physics tests and a total of 85 cycles of 14 NPP units were simulated by TORCH V2.0. However, the content of the reactor startup physics tests varies in different cycles.

Reactor Startup Physics Test

Compared parameters of the reactor startup physics test include critical boron concentration (CBC), control rod integral value, boron differential value, and isothermal temperature coefficient.

The results of critical boron concentration are shown in **Figure 3**. The absolute errors of the critical boron concentration in the all-rod-out (ARO) state are within ±50 ppm and those of the rod-inserted state agreed with the rod-inserted criterion. The formula of the rod-inserted criterion is as follows:

$$(\mathrm{CBC})_{\mathrm{R}}^{\mathrm{M}} = (\mathrm{CBC})_{\mathrm{R}}^{\mathrm{P}} + \left[(\mathrm{CBC})_{\mathrm{ARO}}^{\mathrm{M}} - (\mathrm{CBC})_{\mathrm{R_0}}^{\mathrm{M}}\right] \pm f\left((\mathrm{CBC})_{\mathrm{R_0}}^{\mathrm{M}}, (\mathrm{CBC})_{\mathrm{R}}^{\mathrm{M}}, (\mathrm{CBC})_{\mathrm{R_0}}^{\mathrm{P}}, (\mathrm{CBC})_{\mathrm{R}}^{\mathrm{P}}\right),$$

where M means the measured value, P means the predicted value, R_0/R means the initial/final rod position at the time of measurement, and f is the value of the uncertainty of the measurement, including chemical analysis uncertainty.

The results indicate that the accuracy of ARO calculation is better than that of rod-inserted calculation with the exception of a

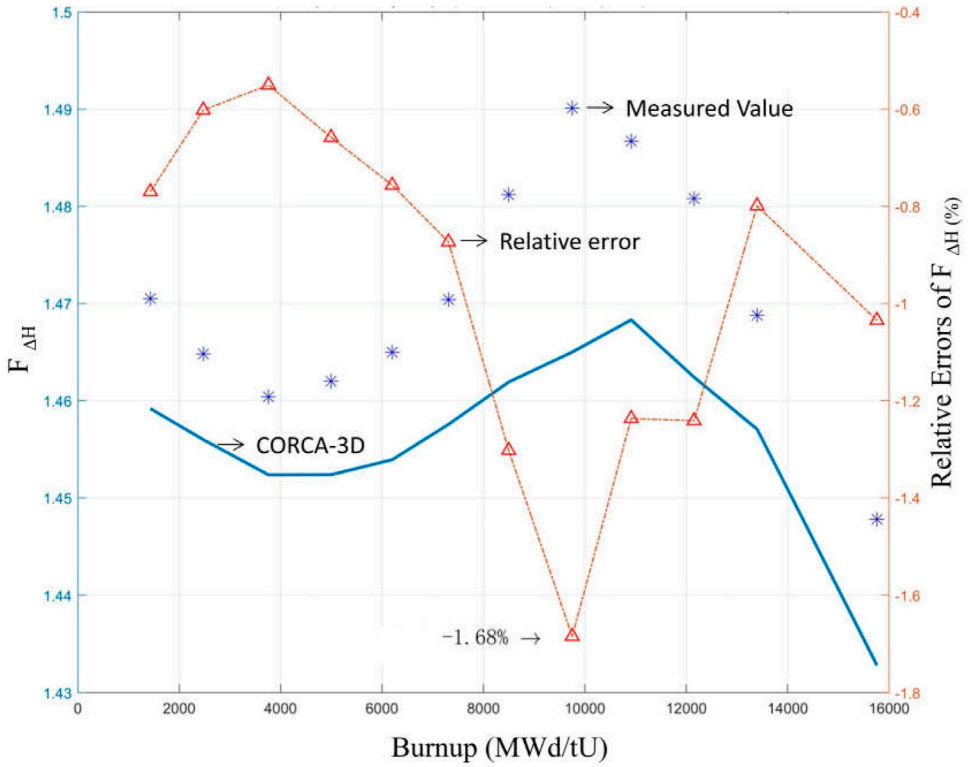

FIGURE 14 | Hot spot factors of Daya Bay NPP (Unit 1 Cycle 11).

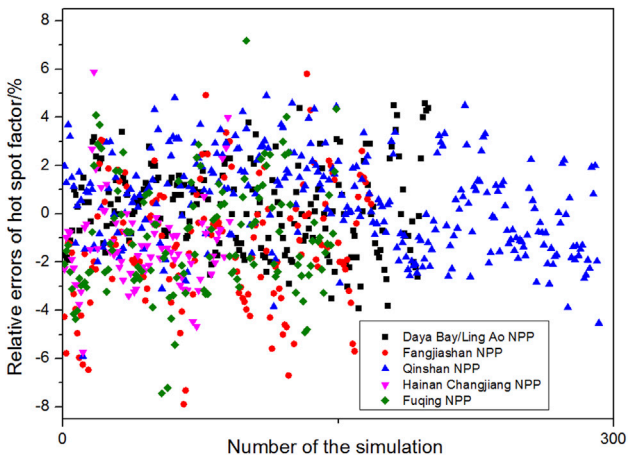

FIGURE 15 | Results of the hot spot factor.

very few calculations. It is because when the control rods are inside the core, the effect and shadow effect would increase the difficulty for the core simulation.

The results of control rod integral value, boron differential value, and isothermal temperature coefficient are shown in **Figure 4**, **Figure 5**, and **Figure 6**, respectively. The relative errors of control rod integral value are within 10%, the absolute errors of boron differential value are within 1.0 pcm/ppm, and the absolute errors of isothermal temperature coefficient are within 3.6 pcm/°C. The large errors appearing in the results belong to the subsequent cycles. It is caused by the core historical effect. In CORCA-3D, the micro burnup method is applied to handle this problem. The models which are based on the two-step calculation scheme adopted in the software cannot fully account for the historical processes. Anyway, all of them agreed with the acceptance criteria. It is indicated that TORCH V2.0 software has enough accuracy in predicting the reactor startup physics test parameters.

Nuclear Power Plant Operation

Compared parameters of the NPP operation include critical boron concentration, assembly-wise power distribution, hot spot factor, and nuclear enthalpy rise factor. The partial results of critical boron concentration are shown in **Figure 7**. The absolute errors of critical boron concentration of all the measured points shown in **Figure 8** are within ±50 ppm except for very few points. At the beginning and end of the core cycle life, the absolute errors are almost worse than those of other times. The changes in xenon concentration and distribution at the beginning of the core cycle life and the core historical effect enhanced with depletion lead to increased errors.

The partial results of maximum assembly-wise power relative error are shown in **Tables 2**–**4**, summary of the maximum assembly-wise power relative errors of all burnup steps is shown in **Figure 9** and **Figure 10**, and the maximum relative errors are within 5% when the relative assembly power is greater than 0.9 and within 8% when it

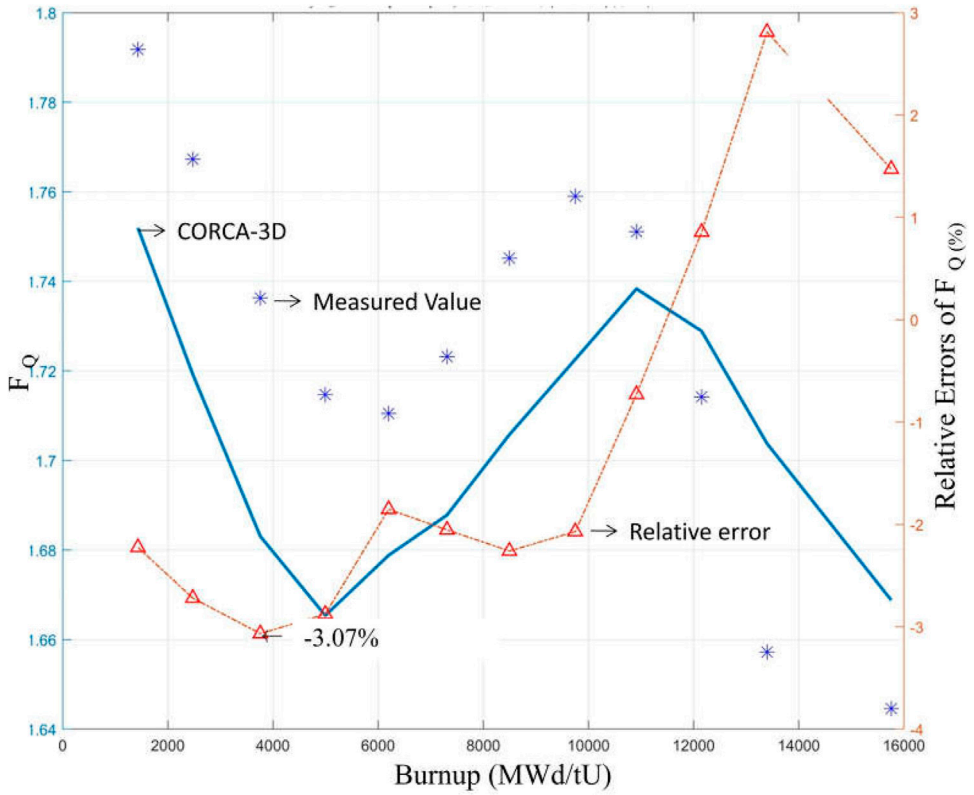

FIGURE 16 | Nuclear enthalpy rise factors of Daya Bay NPP (Unit 1 Cycle 11).

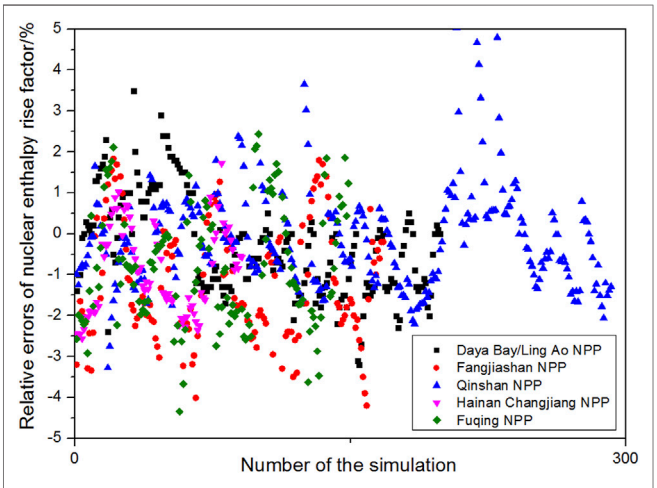

FIGURE 17 | Results of the nuclear enthalpy rise factor.

is less than 0.9, except for very few points. The assembly-wise power distributions of Daya Bay NPP (Unit 1 Cycle 11) at the beginning, middle, and end of the core life are shown in **Figures 11–13** respectively. It can be found that the calculation accuracy does not get worse with the increase in depletion.

The hot spot factors of Daya Bay NPP (Unit 1 Cycle 11) are shown in **Figure 14**. The relative errors of hot spot factors of all cycles are shown in **Figure 15**; the relative errors of hot spot factor are within 8%. The nuclear enthalpy rise factors of Daya Bay NPP (Unit 1 Cycle 11) are shown in **Figure 16**. All the relative error results of nuclear enthalpy rise factor are shown in **Figure 17**; the relative errors are within 5%.

All the calculated parameters are in good accordance with the measured values, which are in agreement with the acceptance criteria. For each core cycle of the NPP used to perform the validation, the average computer processing time of calculations is several minutes. Before the core simulation, the assembly-homogenized few-group constants and the databank of core front-order loops should be ready. The results show that TORCH V2.0 software has high calculation accuracy and efficiency for each cycle operation data of each power plant.

CONCLUSION

In this article, the main components of the PWR core design software package TORCH V2.0 and the main theoretical models and program features adopted in different codes are briefly introduced, and TORCH V2.0 software is verified by using the data of the reactor startup physics test and NPP operation of each power plant. With the exception of very few results of critical boron concentration for some deep operation

cycles, all the results of the compared parameters are in good accordance with the measured values, which are in agreement with the industrial acceptance criteria. The results show that the software TORCH V2.0 has reliable calculation ability and can be applied in the PWR nuclear power engineering design which is based on square fuel assembly. In the future, the uncertainty analysis of TORCH V2.0 would be carried out for the comprehensive verification and validation.

DATA AVAILABILITY STATEMENT

The raw data supporting the conclusions of this article will be made available by the authors, without undue reservation.

AUTHOR CONTRIBUTIONS

XP contributed significantly to analysis and manuscript preparation; CZ performed the data analysis; WZ contributed to analysis and manuscript preparation; QL performed the data analyses and wrote the manuscript. All authors contributed to the article and approved the submitted version.

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