



Article

Validation of the SCALE/Polaris–PARCS Code Procedure with the ENDF/B-VII.1 AMPX 56-Group Library: Boiling Water Reactor [†]

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Abstract: The SCALE/Polaris–PARCS code procedure has been used in the confirmatory analysis for boiling water reactors by the US Nuclear Regulatory Commission. In this study, the SCALE/Polaris v6.3.0–PARCS v3.4.2 code procedure with the Evaluated Nuclear Data File (ENDF)/B-VII.1 AMPX 56-group library was validated by comparing the simulated results with the measured data for operating boiling water reactors, including Peach Bottom Unit 2 cycles 1–3, Hatch Unit 1 cycles 1–3, and Quad Cities Unit 1 cycles 1–3. The uncertainties and biases of the SCALE/Polaris–PARCS code package for boiling water reactor physics analysis were evaluated in the validation for key nuclear parameters such as reactivity and traversing in-core probe data.

Keywords: boiling water reactor; SCALE/Polaris; PARCS; validation; ENDF/B-VII.1; AMPX 56-group library



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1. Introduction

The SCALE/TRITON [1] 2D lattice physics code developed by Oak Ridge National Laboratory (ORNL) has been used in the past to generate few-group assembly- and reflector-homogenized cross sections for 3D whole-core nodal diffusion calculations for light-water reactor (LWR) analysis using PARCS [2], which was developed by the University of Michigan (UM). SCALE/TRITON is SCALE's general reactor physics sequence that couples neutron transport and depletion. SCALE/TRITON uses SCALE/XSProc [3] to obtain effective self-shielded cross sections for the neutron transport calculation, and it calls on SCALE/ORIGEN for depletion. The SCALE/TRITON–PARCS code procedure was used in the confirmatory analysis conducted by the US Nuclear Regulatory Commission (NRC) for LWRs.

Because the XSProc point-wise slowing-down calculation is very time consuming, a new 2D lattice physics code, SCALE/Polaris [4,5], was developed for LWR analysis to provide better computational efficiency and accuracy. Polaris uses the embedded self-shielding method (ESSM) [6] to obtain effective self-shielded cross sections for eigenvalue transport calculation. The ESSM is a variation of the Bondarenko resonance self-shielding method and directly uses the SCALE/AMPX multigroup (MG) library [7] to obtain effective self-shielded cross sections from the resonance cross-section tables as a function

of background cross sections for each nuclide. To enhance the accuracy of the Polaris calculations, resonance cross-section tables in the AMPX MG libraries are generated using selectively heterogeneous or homogeneous models or narrow resonance approximation for all the Evaluated Nuclear Data File (ENDF)/B library nuclides.

At the time of writing, the SCALE/Polaris–PARCS two-step procedure is the NRC’s primary confirmatory analysis tool for boiling water reactors (BWRs) and pressurized water reactors (PWRs). Nuclear vendors and institutions have developed their own two-step code packages for in-core fuel management and safety analysis which are validated to obtain uncertainties for key nuclear parameters by comparing the benchmark results for operating LWRs with plant-measured data. As they improve the two-step code packages with the new version of the ENDF/B nuclear data library, they repeat validations to update the uncertainties for key nuclear parameters to be used in nuclear design.

This study focuses on validating the SCALE/Polaris v6.3.0–PARCS v3.4.2 code procedure for BWRs with the ENDF/B-VII.1 AMPX 56-group library by evaluating uncertainties for reactivity and traversing in-core probe (TIP) data prediction for neutron flux distribution. In addition, the limitations and drawbacks of the SCALE/Polaris–PARCS code procedure are identified for BWR analysis. Four BWRs were selected for which the design, operational, and measured data are publicly available through the Electric Power Research Institute (EPRI) technical reports: Peach Bottom Unit 2 (PB2) cycles 1–3 [8,9], Hatch Unit 1 (Hatch1) cycles 1–3 [10,11], Quad Cities Unit 1 (QC1) cycles 1–3 [12,13], and Monticello cycles 1–3 [14]. However, because Monticello cycles 1–2 include burnable poison (BP) curtains and simplified Polaris–PARCS models introduce unexpectedly large reactivity bias, Monticello was excluded from the validation.

Section 2 summarizes the SCALE/Polaris–PARCS code procedure, including GenPMAXS [15] for cross-section functionalization and PATHS [16] for the BWR thermal-hydraulic feedback. The benchmark results are summarized and uncertainties for key nuclear parameters are given in Section 3. The Discussion and Conclusions are presented in Sections 4 and 5, respectively.

2. Benchmark Calculations

2.1. SCALE 6.3/Polaris–PARCS Procedure

Detailed reactor physics analysis for PWR and BWR has been challenging because it requires significant computing capacity due to the complex geometry and the large size of the reactor core. The most reliable reactor physics analysis for LWR can be performed using continuous energy (~100,000 energy points) Monte Carlo neutron transport calculations with multi-physics analysis, including neutron transport, thermal-hydraulic feedback, and depletion. Because this reactor physics analysis requires high-performance computing using super-computers, it is not practical at all.

A more practical and reliable analysis can be achieved by performing direct 3D deterministic transport calculations with multi-physics using a code package such as the Virtual Environment for Reactor Application (VERA) [17] with a coarse-group (~50 groups) neutron cross-section library. The number of energy groups is significantly reduced compared to continuous energy Monte Carlo calculation, but there is almost no approximation in geometry, which requires much less computing capability, making it more practical. However, because this approach still requires several thousand processors to simulate PWR or BWR cores using a large-sized cluster, it may not be practical, and a more practical approach is necessary.

The most practical analysis procedure is the two-step procedure, for which only serial calculations are performed. Nuclear vendors have developed two-step code packages for LWR analysis. The two-step procedure includes the following: (a) 2D neutron transport calculations for single assemblies and reflectors adjacent to a single fuel assembly to obtain multiplication factors and neutron flux distributions. Then, assembly- or reflector-homogenized few-group (2 or more) neutron cross sections are obtained by flux-volume weighting and energy group collapsing. (b) Three-dimensional few-group

nodal diffusion calculations for the whole core are performed using the assembly- and reflector-homogenized few-group cross sections obtained by the first step (a).

Assembly- and reflector-homogenized few-group cross sections and power form factors to be used in pin power reconstruction are obtained by performing 2D lattice transport calculations using Polaris with the ENDF/B-VII.1 AMPX 56-group library and then flux–volume weighting with energy groups collapsing from 56 groups to 2 groups. Figure 1 demonstrates the representative Polaris single assembly model with a reflecting boundary condition and the Polaris radial reflector model adjacent to a single assembly to obtain assembly- and reflector-homogenized 2-group cross sections. Various reactor states are considered in depletion and branch calculations. Then, few-group cross sections are tabulated using GenPMAXS as a function of various reactor states such as burnups, moderator densities, moderator and fuel temperatures, and control rod insertions. Once cross-section table sets are generated for fuel assemblies with axial heterogeneities and radial and axial reflectors, 3D whole-core diffusion calculations are performed using PARCS. Figure 2 is a flowchart of the Polaris–PARCS procedure for BWR analysis in which two-phase thermal–hydraulic feedbacks are considered using PATHS. Table 1 provides the six reference cases for each type of fuel assembly, and each reference case includes 24 branch states, as provided in Table 2.

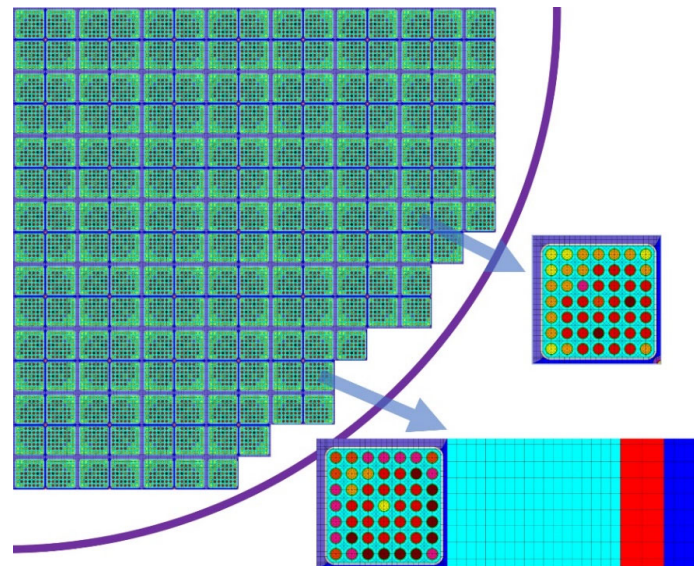


Figure 1. Reactor physics analysis with two-step procedure for BWRs.

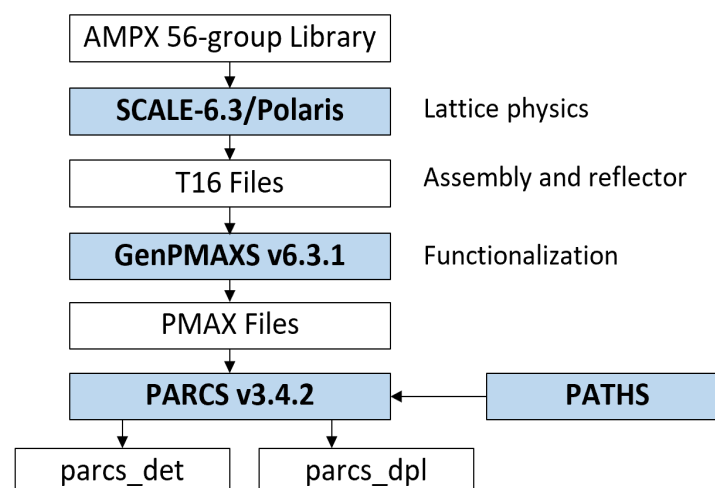


Figure 2. Flowchart of the Polaris–PARCS procedure for BWRs.

Table 1. Reference cases for each assembly type.

Case	Coolant		Moderator		Temperature (K)					CR in
	Density (g/cm ³)	Temp. (K)	Density (g/cm ³)	Temp. (K)	Fuel	Clad	Gap	CR ^a	Channel	
A1	0.7373	560.29	0.7373	560.29	948.45	637.92	793.19	560.29	560.29	no
A2	0.4573	560.29	0.7373	560.29	948.45	637.92	793.19	560.29	560.29	no
A3	0.2473	560.29	0.7373	560.29	948.45	637.92	793.19	560.29	560.29	no
A4	0.7373	560.29	0.7373	560.29	948.45	637.92	793.19	560.29	560.29	yes
A5	0.4573	560.29	0.7373	560.29	948.45	637.92	793.19	560.29	560.29	yes
A6	0.2473	560.29	0.7373	560.29	948.45	637.92	793.19	560.29	560.29	yes

^a Control rod.**Table 2.** Branch states for each reference case.

No.	Coolant		Moderator		Temperature (K)					CR in
	Density (g/cm ³)	Temp. (K)	Density (g/cm ³)	Temp. (K)	Fuel	Clad	Gap	CR	Channel	
1	0.7373	560.29	0.7373	560.29	948.45	637.92	793.19	560.29	560.29	no
2	0.4573	560.29	0.7373	560.29	948.45	637.92	793.19	560.29	560.29	no
3	0.2473	560.29	0.7373	560.29	948.45	637.92	793.19	560.29	560.29	no
4	0.7373	560.29	0.7373	560.29	948.45	637.92	793.19	560.29	560.29	yes
5	0.4573	560.29	0.7373	560.29	948.45	637.92	793.19	560.29	560.29	yes
6	0.2473	560.29	0.7373	560.29	948.45	637.92	793.19	560.29	560.29	yes
7	0.1073	560.29	0.7373	560.29	948.45	637.92	793.19	560.29	560.29	no
8	0.1073	560.29	0.7373	560.29	948.45	637.92	793.19	560.29	560.29	yes
9	0.7373	560.29	0.7373	560.29	500	548.23	524.12	560.29	560.29	no
10	0.4573	560.29	0.7373	560.29	500	548.23	524.12	560.29	560.29	no
11	0.2473	560.29	0.7373	560.29	500	548.23	524.12	560.29	560.29	no
12	0.1073	560.29	0.7373	560.29	500	548.23	524.12	560.29	560.29	no
13	0.7373	560.29	0.7373	560.29	500	548.23	524.12	560.29	560.29	yes
14	0.4573	560.29	0.7373	560.29	500	548.23	524.12	560.29	560.29	yes
15	0.2473	560.29	0.7373	560.29	500	548.23	524.12	560.29	560.29	yes
16	0.1073	560.29	0.7373	560.29	500	548.23	524.12	560.29	560.29	yes
17	0.7373	560.29	0.7373	560.29	1500	748.23	1124.12	560.29	560.29	no
18	0.4573	560.29	0.7373	560.29	1500	748.23	1124.12	560.29	560.29	no
19	0.2473	560.29	0.7373	560.29	1500	748.23	1124.12	560.29	560.29	no
20	0.1073	560.29	0.7373	560.29	1500	748.23	1124.12	560.29	560.29	no
21	0.7373	560.29	0.7373	560.29	1500	748.23	1124.12	560.29	560.29	yes
22	0.4573	560.29	0.7373	560.29	1500	748.23	1124.12	560.29	560.29	yes
23	0.2473	560.29	0.7373	560.29	1500	748.23	1124.12	560.29	560.29	yes
24	0.1073	560.29	0.7373	560.29	1500	748.23	1124.12	560.29	560.29	yes

Typically, the two-step procedure for BWR analysis introduces much larger errors or biases compared to the two-step procedure for PWR analysis. While PWR reactivity is controlled mostly by soluble boron during normal operation, BWR reactivity is controlled only by control rods. Because few-group cross sections cannot be generated that account for

all the operation history of control rod movements in the BWR, more errors are introduced in the simulation results. In addition, two-phase thermal-hydraulic models and self-shielded cross sections for high voids would introduce more errors and biases.

2.2. Benchmark Calculations

2.2.1. Overview

Benchmark calculations were performed for PB2 cycles 1–3, Hatch1 cycles 1–3, and QC1 cycles 1–3 using the SCALE/Polaris v6.3.0–PARCS v3.4.2 code procedure with the ENDF/B-VII.1 AMPX 56-group library. Detailed design data, operational history, and measured TIP data are provided in the EPRI technical reports EPRI NP-563 [8] and EPRI NP-971 [9] for PB2, EPRI NP-562 [10] and EPRI NP-2106 [11] for Hatch1, and EPRI NP-240 [12] and EPRI NP-552 [13] for QC1. The SCALE/Polaris–PARCS benchmark results were compared with the plant-measured data for reactivity and TIP. Detailed information and benchmark results are provided in the ORNL technical report ORNL/TM-2023/2983 [18] for PB2, ORNL/TM-2023/2991 [19] for Hatch1, and ORNL/TM-2023/3002 [20] for QC1.

It is necessary to estimate pin peaking factor uncertainty for BWRs by performing benchmark calculations for critical experiments or comparing pin power distributions between the PARCS result and the gamma scan data. The EPRI report NP-3568 [21] provides gamma scan data measuring ^{140}La only for one fuel assembly at the ends of cycles 1–3. These data can be used to estimate pin peaking factor uncertainty for BWRs. This evaluation was not performed in this investigation. In addition, the EPRI report NP-561 [22] provides various measured gamma scan data obtained at the end of Hatch1 cycle 1. Even though they do not include pin-wise power distributions, they can be used to validate the assembly power prediction capability of the Polaris–PARCS code procedure.

2.2.2. BWR Benchmark Problems

To perform meaningful benchmark calculations, the BWR core design data, operating history, and measured data must be provided. The BWR core design data include specifications for core configuration, fuel assembly data and loading pattern, orifice configuration, and control bank and TIP detector layouts. The operating history must include core pressure, inlet enthalpy, flow rate, power level, and control bank positions.

The hot full-power (HFP) measured data include the TIP data at the specified burnup points. Publicly available BWR plant data and documents were collected for the Polaris–PARCS benchmark calculations. The EPRI reports provide detailed information for PB2 cycles 1–3, Hatch1 cycles 1–3, and QC1 cycles 1–3. Table 3 provides the characteristics of the plants important for the benchmark calculations.

Table 3. Specifications of the BWR plants.

Parameter	Peach Bottom 2	Hatch 1	Quad Cities 1
Core power (MW_{th})	3293	2436	2511
Core pressure (psia (bar))	1035 (71.36)	1035 (71.36)	1035 (71.36)
Core flow rate (Mlb/h (kg/s))	102.5 (0.012915)	78.5 (0.009891)	94.12 (0.011859)
Core inlet enthalpy (Btu/lb (kJ/kg))	521.3 (1212.54)	523.7 (1218.13)	521.3 (1212.54)
Number of assemblies	764	560	724
Initial core loading (MTU)	144.4	104.7	140.3
Pin lattice configuration	$7 \times 7/8 \times 8$	$7 \times 7/8 \times 8$	$7 \times 7/8 \times 8$
Active fuel length (cm)	365.76	365.76	365.76
Number of fuel rods	47/64	47/64	47/64
Assembly pitch (cm)	30.48	30.48	30.48
Pin pitch (cm)	1.875/1.6256	1.875/1.6256	1.87452/1.6256
Fuel pellet radius (cm)	0.60579/0.5207	0.607/0.521	0.61976/0.52832
Cladding inner radius (cm)	0.62103/0.53976	0.622/0.540	0.63373/0.53975
Cladding outer radius (cm)	0.71501/0.62611	0.716/0.626	0.71501/0.62611
Number of control banks	185	136	177
Number of TIP instruments	43	31	41
Control rod material	B_4C	B_4C	B_4C
Burnable poison material	Gadolinia	Gadolinia	Gadolinia
In-core detector	Fission TIP	Fission/gamma TIP	Fission TIP

2.2.3. BWR Benchmark Calculations

To cover all the possible BWR states, the two-group cross sections were tabulated as a function of burnup, moderator and fuel temperatures, moderator density, and the control rod. Polaris performs depletion calculations for six reference cases with integrated spacer grids to reduce the number of cases needed. To complete the functionalization of the cross-section table sets, various branch calculations should be performed at each burnup point for each reference case. The number of state points for the Polaris transport calculation for each BWR assembly will be as follows:

$$6 \text{ (reference cases)} \times 60 \text{ (burnup points)} \times 24 \text{ (branches)} = 8640.$$

Preparing a cross-section set for each fuel assembly type is very computationally expensive; this process needs to be improved for better computational efficiency. The Polaris calculations are performed for radial, top, and bottom reflectors sided by a single fuel assembly without depletion calculation but with branch calculations for moderator temperature and moderator densities. Typical Polaris fuel assembly and reflector models for BWRs are similar to the PWR cases, as illustrated in Figure 1. Polaris saves all the required information for the nodal diffusion calculation on so-called T16 files. Although the BWR plant, fuel, and core design data are provided at room temperature conditions, the benchmark calculations are performed for the HFP temperature conditions. Therefore, all the geometry and composition data must be thermally expanded at the hot temperatures. However, thermal expansion was not considered for simplicity.

Cross-section table sets are prepared for various types of fuel assemblies and reflectors using GenPMAXS. If there is axial heterogeneity caused by the spacer grid and axial blanket, then different types of cross-section table sets must be generated. Although only one fuel-to-reflector model is used for radial reflector cross sections, various radial reflector cross-section sets are generated according to various fuel-to-reflector configurations in the reactor core.

Whole-core 3D nodal diffusion calculations are performed using PARCS. The PARCS input requires various types of information for core calculation, such as thermal power, inlet temperature, coolant flow rate, fuel assembly loading pattern, 3D geometry specification, control bank and TIP layouts, and cross-section assignment to each node. Nodal diffusion calculations are performed at various given reactor conditions, including reactor power, flow rate, control bank movement, and burnup for depletion.

3. Results

3.1. BWR Benchmark Results

3.1.1. Hot Full-Power Reactivities

The calculated HFP reactivities for PB2, Hatch1, and QC1 are illustrated in Figures 3 and 4 for all cycles and reload cycles, respectively. These data are used in statistical analysis to obtain the uncertainty for reactivity in pcm (one one-thousandth of a percent) defined as the multiplication factor (k_{eff})—1.0. Detailed information and results are provided in the ORNL technical reports for individual BWRs, including ORNL/TM-2023/2983 [18], ORNL/TM-2023/2991 [19], and ORNL/TM-2023/3002 [20].

3.1.2. Hot Full-Power TIP Data

Comparisons of the TIP data are illustrated in Figure 5, which provides root mean square (RMS) errors between the 3D, 2D, and 1D measured and calculated TIP data at the specified burnup points for PB2, Hatch1, and QC1. These RMS errors are used in statistical analysis to obtain the uncertainty for TIP. The standard deviations and uncertainties for the measured TIP data obtained using the measured data at symmetric TIP positions were estimated for PB2, Hatch1, and QC1. Detailed information and results are provided in the ORNL technical reports for individual BWRs, including ORNL/TM-2023/2983 [18], ORNL/TM-2023/2991 [19], and ORNL/TM-2023/3002 [20].

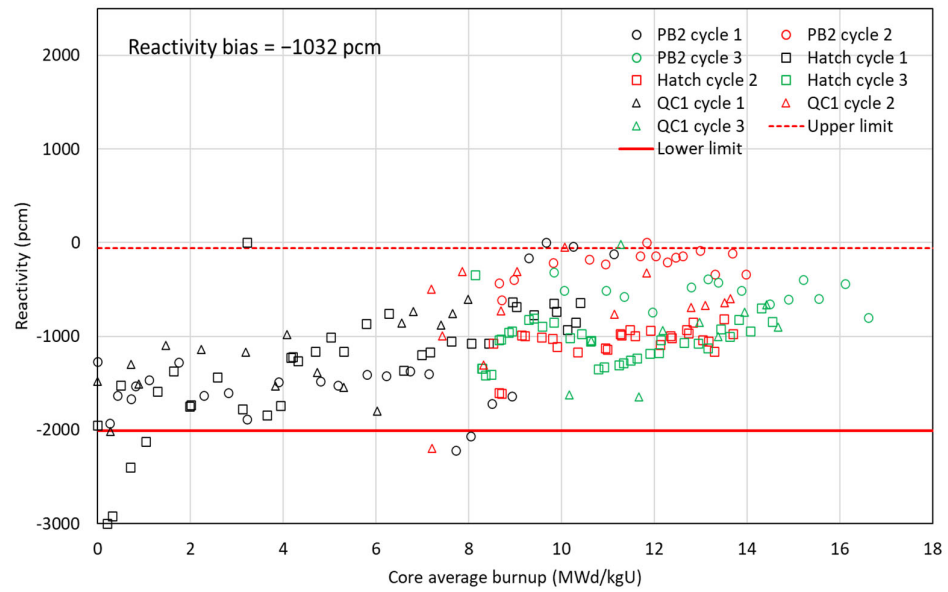


Figure 3. HFP reactivities for all cycles.

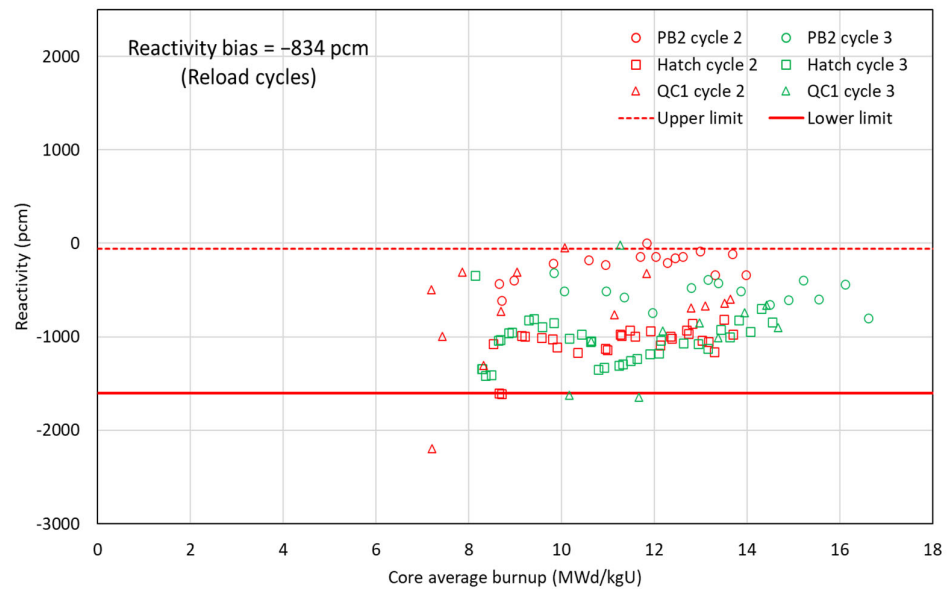


Figure 4. HFP reactivities for reload cycles.

3.2. Uncertainty Evaluation

Statistical analysis was performed using the parametric statistical method for BWRs. Table 4 summarizes the total (S_d), measurement (S_m), and calculation (S_c) standard deviations and degrees of freedom (DOFs) for the total (F_d) and measurement (F_m) on the reactivity and TIP data. Assuming no measurement uncertainty for reactivity, the S_c is equal to S_d . The S_c for TIP can be calculated using

$$S_c^2 = S_d^2 - S_m^2, \quad (1)$$

and the DOF for calculation (F_c) is calculated from the following relationship:

$$S_d^4/F_d = S_c^4/F_c + S_m^4/F_m. \quad (2)$$

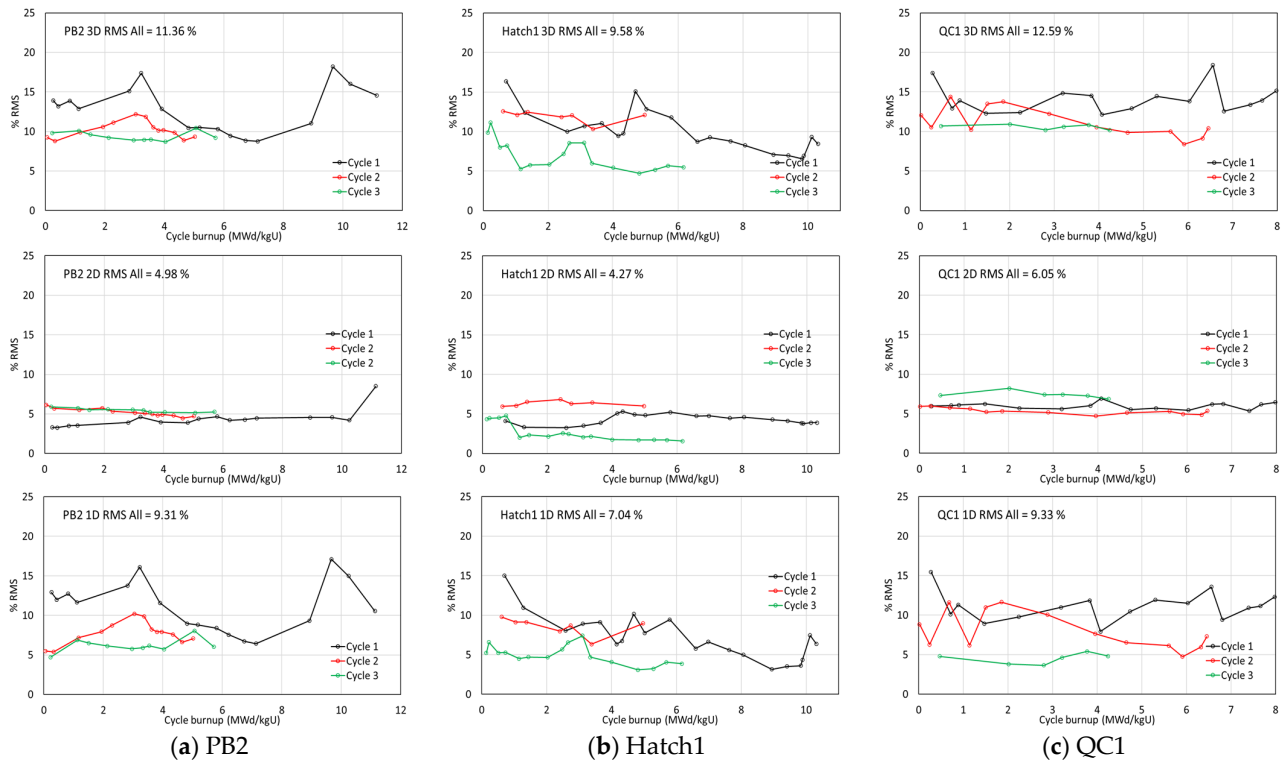


Figure 5. Comparisons of the HFP TIP data for PB2, Hatch1, and QC.

Table 4. Uncertainties for the BWR reactivity and TIP.

Method	Item	Reactivity				TIP		
		All	All Biased	C2 + 3 ^a	C2 + 3 Biased	3D	2D	1D
Parametric statistics (total difference)	S_d	—	—	—	—	11.29%	5.16%	8.56%
	Average cases	—	—	—	—	—	—	—
	$K_{95} \times 95$	—	—	—	—	107,712	4488	2832
	$K_{95} \times 95 S_d$	—	—	—	—	1.645	1.683	1.695
Parametric statistics (measurement)	S_m	—	—	—	—	18.6%	8.7%	14.5%
	Average cases	—	—	—	—	6.08%	3.70%	—
	$K_{95} \times 95$	—	—	—	—	—	—	—
	$K_{95} \times 95 S_m$	—	—	—	—	93,552	3898	—
Parametric statistics (calculation)	S_c	1162 pcm	530 pcm	930 pcm	406 pcm	9.51%	3.59%	8.56%
	Average cases	−1032 pcm	0 pcm	−834 pcm	0 pcm	—	—	—
	$K_{95} \times 95$	194	194	116	116	60,096	1521	2832
	$K_{95} \times 95 S_c$	1.840	1.840	1.904	1.904	1.645	1.712	1.695
Nonparametric statistics	$lower$	2139 pcm	975 pcm	1770 pcm	773 pcm	15.6%	6.2%	14.5%
	$upper$	−2064 pcm	−1032 pcm	−1613 pcm	−779 pcm	—	—	—
Final	$Bias$	−118 pcm	914 pcm	−114 pcm	720 pcm	±15.6%	±6.2%	±14.5%
	$K_{95} \times 95 S$	—	±1032 pcm	—	±779 pcm	—	—	—
	Bias	—	−1032 pcm	—	−834 pcm	—	—	—

^a Cycles 2 and 3.

A normality test was not performed for the HFP reactivities and TIP data assuming normal distribution due to many data points. Uncertainty analysis was performed for burnup-dependent reactivity and TIP data only using parametric statistical analysis. Addi-

tionally, nonparametric statistics analysis for nonnormal distribution was performed for the HFP reactivities using the following equation:

$$P(X_p \leq X_j) = \sum_{k=0}^{j-1} \binom{N}{k} 0.95^k 0.05^{N-k}, \text{ and } X_1 \leq X_2 \leq \dots X_p \leq X_j \leq \dots \leq X_N, \quad (3)$$

where P is a probability, and N denotes the number of data points. When j is found for $P(X_p \leq X_j) \geq 0.95$, the X_j will be a tolerance limit with 95×95 probability and confidence level. Equation (3) could not be used for the TIP data due to a very large number of data points. Table 4 provides the summary for the uncertainty analysis of the BWR key nuclear parameters.

Statistical analysis for reactivity was performed separately for (a) all cycles together and (b) reload cycles only, with and without considering reactivity bias. There are -1032 pcm and -834 pcm reactivity biases for all cycles and reload cycles, respectively. The reactivity biases are too large to neglect and must be considered in analysis. The final reactivity uncertainties were obtained assuming that the reactivity biases should be used in the analysis. The final reactivity uncertainties are ± 1032 pcm for all cycles and ± 779 pcm for reload cycles. Because there are frequent power ups and downs and reactor shutdowns in the initial BWR cycle, a very large reactivity uncertainty is expected. Figures 3 and 4 illustrate the distribution of the reactivity differences between measurement and simulation data for all cycles.

Because many data points for the HFP TIP data are normalized to one, they can be assumed to be a normal distribution. The measurement uncertainty ($K_{95 \times 95} S_m$), where $K_{95 \times 95}$ is a 95×95 one-sided tolerance limit [22], can be calculated using the measured TIP data at the symmetric TIP positions. The calculation uncertainty ($K_{95 \times 95} S_c$) can be calculated using Equations (1) and (2). The final uncertainties for 3D, 2D, and 1D TIP data are $\pm 15.6\%$, $\pm 6.2\%$, and $\pm 14.5\%$, respectively. The uncertainties for 3D and 2D TIP data can be uncertainties for 3D and axially integrated 2D fuel assembly power distributions, which can be combined with the pin peaking factor uncertainty to obtain 3D and 2D pin power uncertainties.

4. Discussion

Axial TIP shape. Figure 6 compares the 1D axial TIP shapes between measurement and simulation for PB2 cycle 2. Because the 1D axial TIP data were obtained by integrating all the radial 2D TIP data at each axial TIP position, it is not easy to obtain the reliable measurement uncertainty for the axial 1D TIP data. However, the 1D TIP measurement uncertainties were estimated to be $\pm 6.4\%$ at 8.7 MWd/kgU and $\pm 5.9\%$ at 13.7 MWd/kgU using Equation (2) with an assumption that 3D standard deviation consists of 1D and 2D standard deviations. These uncertainties can be referred to in the comparison. The overall shapes of the 1D TIP data are roughly consistent, but differences in local shapes might result from neglecting spacer grids in the Polaris-PARCS modeling. Detailed modeling for the seven spacer grids would improve the calculated axial TIP data shapes by providing improved RMS errors. In addition, the two-phase thermal-hydraulic (TH) model might result in differences in TIP data, which require further investigation of the two-phase TH model. Detailed investigation is required to improve the prediction capability for axial and radial power distribution.

Mixed active fuel heights. Fuel rods in Hatch cycle 3 assembly type 5 are axially longer than those in other assembly types by 15.24 cm, requiring mixed axial meshes for nominal and long fuel rods. However, because PARCS cannot model different active fuel heights, these were treated as a nominal fuel rod, resulting in much less fuel mass in the core. This model simplification would result in more negative reactivity and greater uncertainty for the TIP comparison, thus introducing more conservative uncertainties. This issue in PARCS must be resolved because the use of different mixed active fuel heights is a common practice in BWR in-core fuel management.

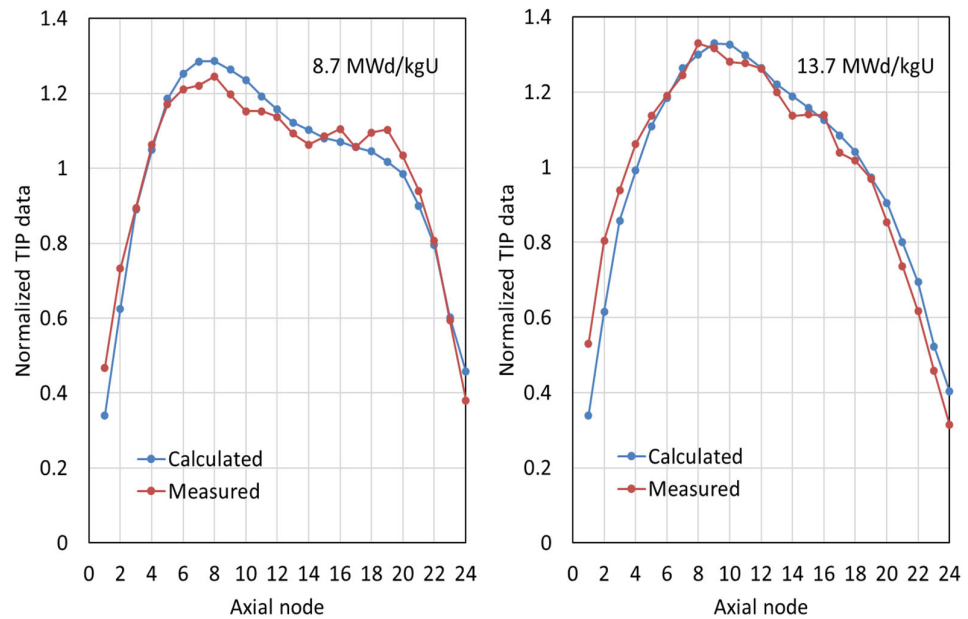


Figure 6. Axial 1D TIP comparison for PB2 cycle 2.

Large reactivity bias for cycle 1. Benchmark results show significant reactivity biases for BWR cycle 1, which might result from very frequent power changes and reactor shut-downs during cycle 1 operation. At the time of writing, all the benchmark calculations rely on the EPRI reports for design data and operating history. The EPRI reports do not always provide detailed operating history information, and sometimes incorrect information or simplified data are provided. The simplified operating history, including frequent reactor shutdowns or power changes, might cause significant reactivity or power differences because of the wrong reactor state in simulation. The large reactivity differences observed in cycle 1 necessitate more detailed investigation. In addition, although some reactivity differences in reload cycles are typical in the two-step analysis, some effort is needed to enhance the overall accuracy by improving cross sections for high-void cases, control rod modeling, and two-phase TH models.

Measured TIP data for Hatch1. EPRI NP-561 notes in Section 4 [23] that the TIP instrument string 26 (D4437) requires “correction factors” by 6 inches (15.24 cm), which is comparable to one axial mesh in the TIP-measured data. Therefore, a sensitivity analysis was performed by adjusting the Hatch1 TIP-measured data for TIP D4437, for which the TIP-measured data were shifted by one, and the last TIP value was taken from the symmetric TIP D3645. A simple comparison of the original and modified RMS errors between the measurement and simulation TIP data shows that the RMS errors were improved with the adjusted measured TIP data for Hatch1 cycles 1 and 2. However, the RMS errors were worsened for Hatch1 cycle 3 as a result of the gamma TIP-measured data. Therefore, uncertainty analysis was performed using the correction factors applied to Hatch1 cycles 1 and 2 measured TIP data and to all Hatch1 TIP-measured data. Table 5 provides comparisons of uncertainties between no adjustment and adjustment through sensitivity analysis. Because there is a difference of only 0.1% in the uncertainty for the 3D TIP data, no adjustment is required when estimating the calculation uncertainty for the TIP prediction.

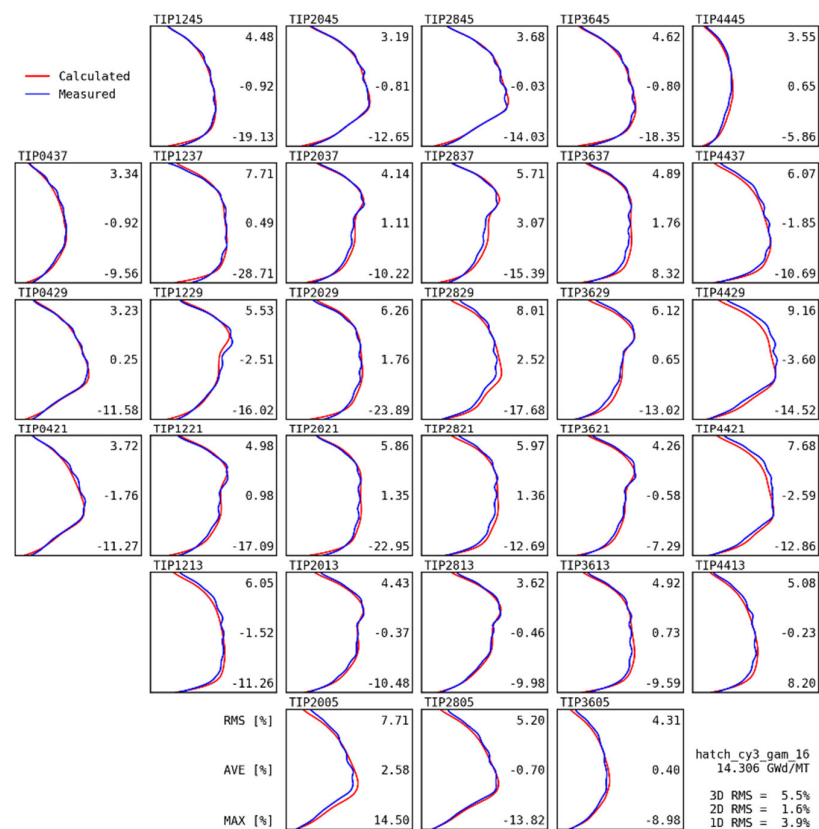
Gamma TIP for Hatch1. The Hatch1 cycles 1–2 and part of cycle 3 include fission TIP, and the remainder of cycle 3 uses gamma TIP. It is known that gamma TIP is more stable than fission TIP. Table 6 compares the standard deviations between the 2D and 3D fission and gamma TIP data for the TIP data differences between the measurement and benchmark and the symmetric measurements. The gamma TIP-measured data show better measurement uncertainty and better agreement with the benchmark results. Figure 7 illustrates a sample comparison of gamma TIP data between the measurement and benchmark results for Hatch1 cycle 3 at a burnup of 14.3 MWd/kgU.

Table 5. Sensitivity analysis with and without adjustment for the Hatch1 TIP-measured data.

Uncertainties		No Adjustment				Adjustment (All Hatch1)				Adjustment (Hatch1 Cycles 1–2)			
		3D	DOFs ^a	2D	DOFs	3D	DOFs	2D	DOFs	3D	DOF	2D	DOFs
Difference	S_d	11.29	107,712	5.16	4488	11.30	107,712	5.16	4488	11.27	107,712	5.16	4488
	$K_{95 \times 95}$	1.645		1.684		1.645		1.684		1.645		1.684	
	$K_{95 \times 95} S_d$	18.6		8.7		18.6		8.7		18.5		8.7	
Measurement	S_m	6.08	93,552	3.70	3898	6.02	93,552	3.70	3898	6.01	93,552	3.70	3898
	$K_{95 \times 95}$	1.645		1.687		1.645		1.687		1.645		1.687	
	$K_{95 \times 95} S_m$	10.00		6.24		9.91		6.24		9.88		6.24	
Calculation	S_c	9.51	60,096	3.59	1521	9.56	60,867	3.59	1521	9.53	60,815	3.59	1522
	$K_{95 \times 95}$	1.645		1.712		1.645		1.712		1.645		1.712	
	$K_{95 \times 95} S_c$	15.6		6.2		15.7		6.2		15.7		6.2	

^a Degrees of freedom.**Table 6.** Comparison between the fission and gamma TIPs for Hatch1.

Uncertainties	Hatch1 Fission TIP				Hatch1 Gamma TIP			
	3D	3D Cases	2D	2D Cases	3D	3D Cases	2D	2D Cases
Difference S_d	10.59%	23,064	4.87%	961	6.25%	8928	2.03%	372
Measurement S_m	6.72%	19,344	3.36%	806	4.23%	7488	2.02%	312

**Figure 7.** TIP comparison for Hatch1 cycle 3 at burnup of 14.3 MWd/kgU.

Thermal expansion. Sensitivity calculation was performed considering thermal expansion for PB2. Figure 8 provides comparisons of the RMS errors with and without consideration of thermal expansion, which are 3D TIP differences between the measurement and benchmark results. A much better agreement is observed for the cycle 1 TIP data with thermal expansion. Figure 9 illustrates the differences in reactivities with and without the consideration of thermal expansion for PB2. The consideration of thermal expansion mostly shows larger reactivities, which would decrease the reactivity biases for BWR.

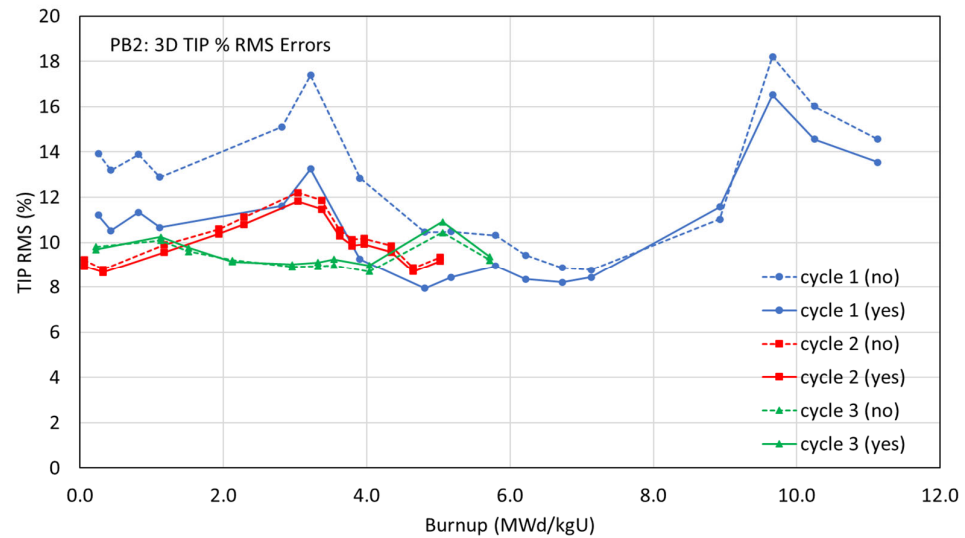


Figure 8. TIP comparison with and without thermal expansion for PB2.

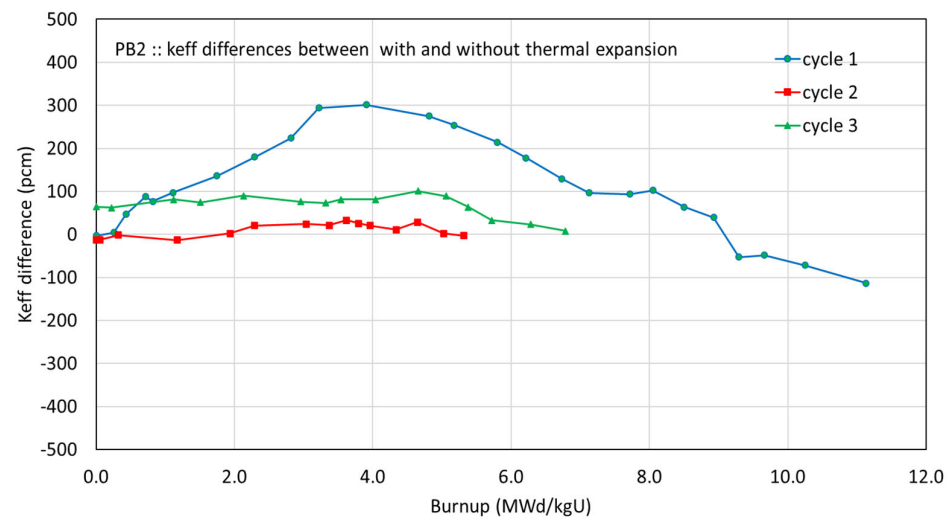


Figure 9. Comparison of the multiplication factors with and without thermal expansion for PB2.

5. Conclusions

The SCALE/Polaris v6.3.0–PARCS v3.4.2 code procedure with the ENDF/B-VII.1 AMPX 56-group library was validated by performing the benchmark calculations for PB2 cycles 1–3, Hatch1 cycles 1–3, and QC1 cycles 1–3. Uncertainties and biases for reactivity and TIP data were obtained by comparing the benchmark results with the plant-measured data and statistical analysis. These uncertainties and biases for BWRs can be used in the NRC confirmatory analysis and in the BWR physics analysis of SCALE and PARCS users.

It is necessary to include more BWR plants with reliable design, operation, and measured data to obtain more reliable uncertainties for the key BWR nuclear parameters. In addition, the SCALE/Polaris–PARCS code procedure and its validation must be further improved for better performance, especially in the following areas:

- Incomplete analysis procedure, especially for radial and axial reflector cross sections;
- Limitation of the SCALE/Polaris–PARCS modeling capability for burnable poison curtain and assembly-dependent active fuel heights;
- Possibility of poor two-phase TH model for BWRs;
- Lack of bypass modeling capability in PARCS;
- Poor reactivity prediction of Polaris with the AMPX 56-group library for very-high-void cases;
- Need to refine Polaris–PARCS models;

- Lack of reliable BWR plant design, operation, and measured data.

The Polaris–PARCS procedure must be improved to provide a standard, robust procedure for users. More LWR plant design and measured data must be collected and included in the validation to be more statistically meaningful.

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