Perspectives on Validation and Uncertainty Evaluation of SFR Nuclear Design Code

Moo-Hoon BAE*, Yong Won CHOI, Andong SHIN, and Namduk SUH
Korea Institute of Nuclear Safety, 62 Gwahak-ro, Yuseong-gu, Daejeon 305-338, Republic of Korea
**Corresponding author: mhbae@kins.re.kr*

1. Introduction

Fast reactors such as PGSFR (Prototype Gen-IV Sodium-cooled Fast Reactor) developed by KAERI have fundamental differences in terms of core characteristics and associated fuel cycle compared to thermal reactors, which need specific new effort for code validation. In current PWRs, nuclear design code systems have been validated using numerous data accumulated by wide operating experience, and its uncertainty can be assessed by statistical methods. However, in order to validate code systems for SFRs with little operating experience, and particularly prototype reactor, new approaches are required.

In this study, a current procedure for validation and uncertainty evaluation is reviewed in nuclear design code systems for PWRs, and global approaches for validation of SFR code systems are surveyed. Through these reviews, perspectives on nuclear design code validation for SFRs are identified.

2. Nuclear Design Code V&V Methodology

“Verification” implies comparisons with reference equation solutions or with analytical solutions, when they exist. In general, reference solutions for verification are provided by Monte-Carlo method such as MCNP, McCARD, etc. “Validation” is based on experiments. In current PWRs, data measured in target reactors are used for code validation. In advanced reactors such as SFRs of Gen-IV type or prototype reactors, two types of validation approaches can and have been used [1]: (1) mock-up experiments, (2) integral experiments. In these experiments, the following measurements and tests are used for code validation: (1) critical measurements, (2) isotopic measurements, (3) power reactor physics tests, (4) power reactor core-follow measurements, (5) power reactor transient measurements, etc.

In this chapter, code V&V and uncertainty evaluation procedure for PWR neutronics simulations are reviewed, and global approach for validation of SFR neutronics simulation codes are surveyed.

2.1 PWRs

The nuclear design code V&V of PWRs is shown in Table 1. In general, nuclear design code systems are divided into two parts: cross-section (XS) generation code, whole-core calculation code. Each code system is verified and validated by each different problem set.

Validation and uncertainty evaluation of total code system including XS generation code and whole-core calculation code are implemented using measured data of target reactors such as WH-type reactor, OPR100, etc.

For code validation, parameters and tests as show in Table 2 are used to validate through comparison between code calculations and measurements.

Table 2. Parameters and Tests for Code Validation

State	Parameters / Tests
Steady	Pin Power distribution - axial/radial power peaking factor Assembly Power distribution - axial/radial power peaking factor Core reactivity and Critical boron density Isothermal temperature coefficient Power coefficient Control rod worth - total/single control rod worth Boron worth, etc.
Transient	SAM (Shape Annealing Matrix) test Xenon oscillation control test Load rejection test Load cycle test Unit load transient, etc.

Table 1. V&V of PWR Nuclear Design Codes

	V&V	Problems	Reference codes
XS generation (A)	Verification	Numerical benchmarks - C5G7 MOX, single fuel pin, 3x3 fuel pins, fuel assembly, color-set problems, etc.	MCNP
	Validation	Critical assemblies: CE, B&W, KRITZ, etc.	CASMO-3, DIT, etc.
Whole-core calculation (B)	Verification	Numerical benchmarks - Steady state: IAEA3D, NEACRP-L336, EPRI-9R, etc. - Transient state : OECE/NEA MOX/UO_2 problems, etc.	VENTURE, PARCS, etc.
A and B	Validation / Uncertainty evaluation	Plant measurements for different fuel cycles - WH-type, OPR1000, etc.	DIT/ROCS, PARAGON/ANC, etc.

Uncertainty of some parameters needed in core operation and safety analysis, among nuclear characteristics parameters in Table 2, is quantified by statistical method with 95% probability and 95% reliability. In order to calculate tolerance limits of nuclear characteristics parameters, first of all, normality of sample group of differences (M-C) between measured values (M) and calculated values (C) is checked by Shapiro-Wilk test [2]. By whether it is normal distribution or not, uncertainty is calculated as follows:

a) normal distribution: statistical method (χ^2 method, etc.) for normal distribution
b) non-normal distribution: more conservative value between statistical method for normal distribution and non-parametric statics

2.2 SFRs

In order to validate and quantify uncertainty on SFR neutronics simulation codes, different approach with existing procedure for PWRs is needed because operating experience isn't many and SFRs have different nuclear characteristics and fuel cycle with PWRs. Two types of validation approaches can and have been used as shown in Table 3.

Table 3. Validation Approaches for SFRs

Experiments	Descriptions
Mock-up exp.	- very close to a target reactor - need of assessment of similarity between models of mock-up experiment and target reactor (critical mass, geometry, spectral index, reaction rate distribution, etc.)
	<u>Examples</u> - BFS (Russia), ZPPR (USA), FCA (Japan), etc.
Integral exp.	- well-documented and well-established experiment - being collected within IRPhEP*
	<u>Examples [1]</u> - Physics experiments at reactor start-up: SuperPhenix - Operation experiments: EBR-II, FFTF, Phenix, JOYO, etc. - Irradiation experiments: PROFIL and TRAPU experiments in Phenix

* International Reactor Physics Experiment Evaluation Project (IRPhEP) [3]

Also, uncertainty evaluation of nuclear characteristics parameters is required for safety analysis. If uncertainty is calculated by similar method to PWRs, the evaluated values will be predicted too conservatively because of insufficient measured data. In order to complement this drawback and reduce the uncertainty, global approaches as follows have been studied [4, 5]: (1) bias factor method, (2) cross-section adjustment method, etc. However, it has to be confirmed whether these method has been approved for use by foreign regulatory bodies.

3. Perspectives on SFR Nuclear Design Code V&V

Based on above review, the following considerations related to SFR nuclear design code V&V and uncertainty evaluation were preliminary identified.

a) Similarity evaluation between mock-up experiment and target reactor: critical mass, geometry, spectral index, reaction rate distribution, etc.
b) Evaluation of reliability on predicted uncertainty, when statistical method is used: normality, number of sample, etc.
c) Detailed evaluation on uncertainty quantification of sodium void worth with large uncertainty and as important safety parameter
d) Conservative uncertainty in initial operating stage, and Re-evaluation of uncertainty using various core physics test in commissioning
e) Detailed establishment of core physics tests for uncertainty re-evaluation

4. Conclusions

In case of neutronics code V&V, current procedure for PWRs and global approaches for SFRs were reviewed and surveyed. Though this review, perspectives on nuclear design code V&V and uncertainty evaluation for SFRs were identified. Further study will be implemented to obtain more insight on code validation.

REFERENCES

[1] G. Palmiotti et al., Validation of Simulation Codes for Future Systems: Motivations, Approach and thje Role of Nuclear Data, *4th Workshop on Neutron Measurements, Evaluations and Applications*, Oct., 2007.
[2] Shapiro, S.S. and Wilk, M.B., An Analysis of Variance Test for Normality (Complete Samples), Biometrika, 52, 3 and 4, p.591-611, 1965.
[3] "International Handbook of Evaluated Reactor Physics Benchmark Experiments", NEA/NSC/DOC, 2006.
[4] T. Takeda et al., Prediction Uncertainty Evaluation Methods of Core Performance Parameters in Large Liquid-Metal Fast Breeder Reactors, *Nuclear Science and Engineering*, **103**, 157-165, 1989.
[5] G. Palmiotti et al., A Global Approach to the Physics Validation of Simulation Codes for Future Nuclear Systems, *Int. Conf. on Reactor Physics, Nuclear Power*, Switzerland, Sep. 14-19, 2008.