Proceedings of the Reactor Physics Asia 2015 (RPHA15) Conference
Jeju, Korea, Sept. 17-18, 2015

Reactor Physics Activities in Nagoya University

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1. Introduction

Major activities regarding to reactor physics research in Nagoya University is summarized in the present paper.

The severe accident in Fukushima Daiichi Nuclear Power Station drastically changes landscapes of nuclear industry and academia in Japan. The reactor physics community in Japan had made serious discussions on the raison d'etre of reactor physics and its contribution to the essential part of nuclear power. During this discussion, we again recognized the meaning of reactor physics – only the reactor physics can explain why it is a nuclear reactor, i.e., mechanism of chain reaction, which is the most fundamental part of nuclear power. In this context, we believe that reactor physics shall be responsible for the essential part of nuclear safety.

The above discussion is driving force for us to promote reactor physics research not only on conventional reactor physics field but also on more nuclear safety oriented field. There might be arguments on the "responsible area" of reactor physics. Now we believe that reactor physics should be "physics of reactors" since most of physical phenomena has impact on behavior of a reactor.

2. Recent Research Activities

2.1 Uncertainty estimation, sensitivity analysis, and cross section adjustment

Nuclear reactor has large potential hazard thus its safety is strictly secured through design calculations based on numerical simulations. Numerical simulations inevitably contains errors and uncertainties due to input data and numerical models. Uncertainty of prediction calculation is usually considered as design margin to guarantee performance and safety of a reactor. Thus uncertainty estimation is an essential part of nuclear safety.

Uncertainty estimation of un-measured core parameters is important to confirm safety. For example, magnitude of the Doppler coefficient is crucial for transient behavior especially in the case of reactivity insertion accident, but it is difficult to measure and confirm uncertainty of the Doppler coefficient. However, we know that errors of many core parameters are highly correlated. It implies that uncertainty (or error) of un-measured core parameters can be estimated through error of measured core parameters using the correlation among errors of core parameters. Based on this idea, errors of control rod bank worth in a typical PWR core is estimated using difference of calculated and measured assembly power distribution. The results indicate that the error of bank worth is well predicted using the difference on assembly power distribution [1].

Uncertainty estimation of LWR reload core parameters are important but has not been investigated except for a few advanced works by Khalik et al [2]. In LWR analysis, a complicated neutronics analysis sequence based on assembly-core two step approach is usually adopted. Thus error propagation from input data (e.g., cross section) to core characteristics is complicated. Furthermore, non-linear effect due to thermal hydraulics and burnup should be explicitly taken into account. For this problem, we tried to estimate uncertainty of reload core parameters based on the random sampling technique in realistic PWR and BWR cores using the licensing grade core simulators [3]. This research reveals several interesting results, e.g., uncertainty of assembly power distribution in BWR is much smaller than that of PWR though uncertainties of k-effective in BWR and PWR are comparable.

Cross section adjustment has been widely used in fast reactor analysis as a practical approach to reduce prediction uncertainty. However, its application to LWRs has been impractical since estimation of sensitivity coefficients requires huge computational resources. The cross section adjustment method based on the random sampling has been developed to make a breakthrough in application of LWR analysis. In the conventional cross section adjustment method, sensitivity coefficients are explicitly estimated since the formula used for cross section adjustment contains the sensitivity matrix. However, in our approach, correlation between perturbation of cross section and perturbation of core characteristics is used instead of the sensitivity matrix [4]. Correlation between perturbations of cross section and core characteristics can be obtained by the random sampling calculation results, thus this approach is very practical since existing core analysis system can be directly applied. Verification results show that the proposed method is a practical candidate for cross section adjustment in LWR analysis.

Major activity in this field is uncertainty estimation of prediction results due to error propagation from input data. This is an important part of uncertainty estimation, but numerical model (e.g. discretization) also contributes to the uncertainty of calculation results. Unfortunately, uncertainty estimation of calculation models is much more difficult and challenging. Key question is: "How to estimate error of an approximate numerical model without a reference solution?" In common verification and validation process, error of numerical simulation is estimated through well-defined benchmark problems. However, no quantitative discussion on its validity in actual reactor condition has been carried out. We propose a primitive framework to predict calculation modeling error using correlation of calculation model errors and some observables (e.g., sensitivity coefficients) obtained in the simplified calculation [5]. The preliminary analysis suggests that the proposed framework would be feasible.

2.2 Critical safety, subcriticality estimation

Criticality safety is a traditional area but various difficult issues still exists mainly due to the present framework of reactor physics that is based on the expansion of eigen-function and focuses on the largest eigenvalue and associate eigen-function. However, in a subcritical system, higher eigen-functions are excited by external source thus their impact should be taken into account to accurate estimate subcriticality. In this context, for example, conventional definition of k-effective would not be appropriate to represent characteristics of a subcritical system. We made various and essential discussions on representative parameters for sub-critical multiplication and a new parameter, k-det, is proposed as a parameter to represent subcriticality [6]. Relation among conventional parameters, i.e., k-effective and k-sub, is also clarified.

Subcriticality of ADS is a very important parameter for performance of ADS thus its monitoring is important. Since proton accelerator of ADS performs pulse mode operation, neutron pulse is periodically injected to a subcritical core of ADS. In this situation, the pulsed neutron method is appropriate to measure subcriticality of ADS. Though the pulsed neutron method seems to be a classical theory, its validity under existence of higher order modes has not been extensively carried out. In actual situation, measured subcriticalities show significant position dependence due to higher order modes. We propose a correction method utilizing correlation between the area ratio and subcriticality based on the Bayes theorem [7].

Subcritical measurement using inherent neutron source of nuclear fuel and the Feynman-α method is also investigated in order to perform subcritical monitoring in various situations such as fuel fabrication factory, fuel debris in Fukushima-Daiichi[8].

2.3 Monte-Carlo method

Reliability of the Monte-Carlo method is crucial since it is often used to produce reference solutions. In this context, statistical uncertainty of the Monte-Carlo method would be an important parameter to represent quality of solution. It is well known that statistical uncertainty of local tally parameters obtained by the Monte-Carlo method may show significant underestimation due to inter-cycle correlations. Many excellent works have been done in this field to quantitatively estimate degree of underestimation. We are trying to theoretically reproduce degree of underestimation of statistical uncertainty using the fundamental theory of power iteration used in the Monte-Carlo code. Our research revealed that degree of underestimation of statistical uncertainty of local fission tally can be accurately reproduced by our theory in a simple calculation conditions, e.g., 1D slab-geometry with pure analog tracking. However we also recognize that degree of underestimation has significant dependence on calculation conditions in the Monte-Carlo calculation, e.g., the analog/non-analog tracking, the fission source generation method, and so on [9]. Effort to improve of our theory is still undergoing.

Fukushima Daiichi accident poses a potential criticality safety issue due to fuel debris. Detail location and formation of fuel debris in Unit 1, 2, and 3 are not known, but it may form lumps distributed in water. Treatment of such complicated geometry with deterministic code would be difficult, but the statistical geometry model adopted in many Monte-Carlo code can treat such configuration. However, our research clarifies that a crucial (but implicit) limitation exist in the current statistical geometry model (STGM) due to approximation in the nearest neighbor distribution and its validity becomes lower for large fuel lumps. In order to address this issue, we propose a new STGM based on the delta-tracking and spatial distribution of packing fraction [10]. The verification result indicate that the improved model show better reproducibility of k-infinity for randomly distributed fuel particle geometry.

2.4 Transport calculation method

Method of characteristics (MOC) has been widely used for core analysis of high fidelity. Its application is extending from static to kinetics analysis.

In the rigorous form of MOC kinetics equation, temporal derivative of angular flux appears. In reactor kinetics calculations, the implicit (or the theta) time-integration scheme is usually adopted due to its stiff nature containing short and long time constants. However, direct application of the implicit time-integration scheme to kinetic MOC calculation requires impractical computational memory since angular fluxes on all ray trace segments should be stored on memory. Thus isotropic approximation has been widely used for the temporal derivative of angular flux. In order to address this issue, we propose the on-flight method, in which angular fluxes at the previous time-step are recalculated [11].

Verification calculation of the MOC kinetic calculation method using the on-flight method well reproduce the reference result. Furthermore, the conventional isotropic approximation for the temporal derivative term of angular flux gives sufficiently accurate result for common kinetics calculation, e.g., the LRA benchmark problem. During the course of this study, efficient acceleration scheme, the multi-grid amplitude function method, is also applied to MOC kinetics calculation [12].

Various cutting edge works have been carried out on enhancement of MOC to three-dimensional geometry.

For example, Joo et al. have developed well known core analysis systems, DeCART and nTracer, using the planar MOC method, which is an efficient algorithm to handle neutron transport in heterogeneous three-dimensional geometry [13].

As one of the "true" three-dimensional MOC method, Axially Simplified MOC in 3D (ASMOC3D) method has been developed [14]. Based on the ASMOC3D method, we propose the Legendre Angular Flux Expansion (LEAF) method [15]. In conventional MOC in 2D, sets of characteristics lines are drawn to cover a system. In the LEAF method, instead, the sets of characteristics planes are drawn to cover a three-dimensional geometry. Neutron transport in each characteristics planes are rigorously carried out based on analytic solution of the Boltzmann transport equation. Angular fluxes and neutron source distributions in a characteristics planes are spatially expanded by the Legendre function. The LEAF method can eliminate the major approximation in the planar MOC method and gives equivalent solution with true 3D MOC method. Verification results indicate accuracy and effectiveness of the LEAF method.

Discontinuity factor is widely used for diffusion calculation with the advanced nodal method. However, its application to transport calculation is still limited since discontinuity factor for transport calculation has not been well established. We have proposed discontinuity factor for integro-differential transport equation [16][17].

Isotropic scattering approximation with transport correction has been widely used in MOC calculations. However, a MOC calculation with transport correction would show numerical instability especially for the case of thick water region and fine energy groups. This issue has been recognized so far in the community, but no essential countermeasure has been taken. The reason for numerical instability was pursued and it is found that absolute value of the scattering ratio sometime exceeds unity, since negative self-scattering cross section becomes significantly large for water region in the case of fine energy groups. Under-relaxation for update of scattering source is effective for the instability [18].

2.5 Core analysis method

Pin-by-pin core analysis method has been used for PWR. However, its application to BWR has not been realized due to complicated structure of fuel assembly and thermal hydraulics effects. We have tried to address these issues. At first, treatment of staggered mesh appeared in BWR core analysis investigated based on the SP3 method. Then, whole-core pin-by-pin sub-channel analysis method is investigated in order to reduce its computation time [19]. We also perform investigation on treatment cell-homogenized cross sections for pin-by-pin core analysis, focusing on accurate cross section reconstruction model and reduction of energy-collapsing and homogenization errors [20].

2.6 Computational Science

Core simulator is very useful for educational purpose. However, in a full-scope simulator, response of reactor can be monitored through a control panel, e.g., trend graph of source/power range monitors.

In order to make intuitive understanding of reactor kinetics behavior, a simple core simulator using the augmented reality (AR) technology has been developed. In this simulator, a small reactor model (of desk top size) is prepared and a user can withdraw or insert control rod by his/her hand. Control rod position in the reactor model is recognized through a CCD camera and core kinetics calculation is carried out in real time. Then the calculation results (e.g., power distribution) is shown over the CCD picture of reactor model. Thus a user can "see" power distribution inside a reactor model through "augmented" graphics on the reactor model and its response to control rod maneuver [21].

Fast computation using GPU is getting wide range of attention due to its superiority on computational performance. However, it has different hardware architecture from CPU and thus different programming model is necessary to fully use the capability of GPU. Basically, GPU performs massively parallel processing using many stream processors in it. However, since the cache size is very limited, several special cares are necessary during implementation of a simulation code. We have developed prototype codes for diffusion and Monte-Carlo calculations. Results indicate that performance of GPU is typically several dozen to several times higher than that of CPU [22][23].

2.7 Nuclear safety

There is no symptom of re-critical event for fuel debris in Fukushima Daiichi, there might be a potential risk of re-critical event during removal of fuel debris. Thus rough estimation of consequence occurred by re-critical event would be important for the risk management.

A hypothetical condition for fuel debris is considered and a kinetics calculation with thermal-hydraulics feedback and heat transfer inside a fuel debris or to surrounding water is carried out. The result indicates that the maximum power during re-critical event relies on reactivity insertion rate due to strong negative feedback effect.

Understanding of essence of severe accident progression is very important to effectively perform severe accident management. Progress of severe accident is usually analyzed by dedicated severe accident codes such as MAAP or MELCOR. Since these severe accident codes are very complicated and large, it is very difficult to grasp physical essence of severe accident progression only by analysis by these codes. Thus we have developed a simple and intuitive prediction method of a typical severe accident progression for the use of educational purpose [24]. Only "paper-and-pencil" are necessary to perform prediction of severe accident progression using the present method. A scenario based on the AC/DC station blackout, ultimate loss of cooling capability and heat sink is assumed for a typical PWR and progression of accident is analyzed. The result indicates that the simplified prediction method well reproduces the reference results obtained by MAAP, e.g., timing of major events during accident (e.g. core melt, rupture of reactor vessel, containment vessel), within the error of ±20%.

3. Summary

In the present summary, recent activities on reactor physics in Nagoya University is summarized. We would like to promote reactor physics researches keeping the following fact in mind: "Safety is the first priority in nuclear power".

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