Annals of Nuclear Energy 85 (2015) 501–511

Contents lists available at ScienceDirect

Annals of Nuclear Energy

journal homepage: www.elsevier.com/locate/anucene

Development of a sensitivity and uncertainty analysis code for high temperature gas-cooled reactor physics based on the generalized perturbation theory

Tae Young Han, Hyun Chul Lee *, Jae Man Noh

Korea Atomic Energy Research Institute, 989-111, Daedeok-daero, Yuseong-gu, Daejeon 305-353, Republic of Korea

ARTICLE INFO

Article history:
Received 6 February 2015
Received in revised form 5 June 2015
Accepted 6 June 2015
Available online 16 June 2015

Keywords:
Sensitivity
Uncertainty
Generalized perturbation theory
MUSAD
DeCART

ABSTRACT

A computer code, MUSAD was developed for the uncertainty and sensitivity analysis on the DeCART neutron transport calculations for a high temperature gas-cooled reactor. MUSAD is based on a deterministic method in which the sensitivity coefficients of the multiplication factor and the microscopic cross sections are derived using the generalized perturbation theory. Then, the uncertainties of the reactor physics responses are calculated by the product of the covariance matrix and the sensitivity coefficients. MUSAD has been verified against the uncertainty analyses on several benchmark problems including the GODIVA benchmark problem, the PMR-200 pin-cell, and the MHTGR-350 core benchmark problems. A good agreement in comparison with the reference codes, TSUNAMI and McCARD shows the applicability of MUSAD to the uncertainty and sensitivity analyses on the HTGR neutron transport calculations.

© 2015 Elsevier Ltd. All rights reserved.

1. Introduction

Sensitivity and uncertainty analysis (SUA) is an emerging issue in neutronic analysis of a reactor core since the uncertainty quantification of the core physics parameters is very important in design and safety analysis of a reactor and many researchers have been investigated (Rearden, 2009a,b; Zwermann et al., 2014; Kodeli, 2001; Takeda et al., 2006; Pusa, 2012; Shim and Kim, 2012). Besides, international cooperative projects are conducted actively. The light water reactor (LWR) uncertainty analysis in modeling (UAM) benchmark activity by OECD/NEA expert group (Ivanov et al., 2007; Bratton et al., 2014) has been on-going since 2006 and the International Atomic Energy Agency (IAEA) coordinated research project (CRP) on high temperature gas-cooled reactor (HTGR) UAM started in 2012 (IAEA CRP, 2014).

There are two kinds of approaches in uncertainty quantification: statistical and deterministic approaches. In case of the statistical approach, the statistical errors of the core physics parameters are quantified from the output parameter sets evaluated by using sufficient number of input parameter sets sampled according to the input parameter distribution functions. The XSUSA code adopted this approach and Zwermann analyzed the nuclear data sensitivity and uncertainty of fuel assembly depletion with XSUSA code (Zwermann et al., 2014). In the deterministic approach, on the other hand, the uncertainties of the core physics parameters are quantified based on the generalized perturbation theory (GPT) (Childs, 1980; Ronen, 1986). The TSUNAMI code (Rearden, 2009a,b) in the SCALE code system adopts the deterministic approach. Takeda also has developed SAGEP (Takeda et al., 2006) code based on GPT and Pusa incorporated SUA within the framework of CASMO (Pusa, 2012) using the method. Shim implemented SUA in the McCARD code based on the Monte-Carlo perturbation theory (Shim and Kim, 2012).

Very high temperature reactor (VHTR) is being developed at Korea Atomic Energy Research Institute (KAERI) for hydrogen production application (Chang et al., 2007). The core of a VHTR consists of graphite moderator and the fuel compact which is a mixture of graphite matrix and the TRISO particle fuel, which introduces a unique neutronic characteristic, the double heterogeneity. A two-step neutronic analysis procedure for VHTR core using DeCART (Deterministic Core Analysis based on Ray Tracing) code (Cho et al., 2008) and CAPP (Core Analyzer for Pebble and Prism type VHTRs) code (Lee et al., 2014) is being developed at KAERI. In the first step of the procedure, the DeCART code is used as a lattice physics code to generate few-group constants to be used in the second step. The double heterogeneity of VHTR is treated in the DeCART code by solving the renewal equation in a stochastic medium (Sanchez, 2004). In the second step of the procedure, the CAPP code is used as a core simulation code to generate

* Corresponding author.
E-mail address: lhc@kaeri.re.kr (H.C. Lee).

http://dx.doi.org/10.1016/j.anucene.2015.06.005
0306-4549/© 2015 Elsevier Ltd. All rights reserved.

core physics parameters based on the neutron diffusion theory. The SUA of the DeCART/CAPP code system is also performed in two steps. In the first step of the SUA, deterministic approach based on the perturbation theory is adopted and the uncertainties of the few-group constants generated by the DeCART code are quantified. In the second step, statistical approach is adopted and the uncertainties of the core physics parameters are quantified stochastically. After sampling sufficient number of few-group constant sets sampled according to the uncertainties of the few-group constants generated in the first step, each input set is evaluated by CAPP code and the statistical errors of the core physics parameters are evaluated. A similar approach was first employed by Yankov et al. (2012). MUSAD (Modules for Uncertainty and Sensitivity Analysis for DeCART) code was developed for the first step of the SUA. The MUSAD code can calculate the uncertainty of the multiplication factor based on the classical perturbation theory (CPT) as standalone type and also produce sensitivity coefficients and quantifies the uncertainties of the few-group constants based on GPT using the few-group constants and the generalized adjoint solutions by DeCART.

This paper presents the methodologies and the verification results MUSAD code. The methodologies for SUA implemented in the MUSAD code will be described in Section 2 and the sensitivity and uncertainty results of the MUSAD code for GODIVA benchmark (OECD/NEA, 1995), PMR-200 pin cell problem, and MHTGR-350 benchmark (IAEA CRP, 2014) will be verified by comparing them with the results of the reference codes, TSUNAMI of SCALE 6.1 and McCARD, in Section 3.

2. Methodologies

Uncertainty of an estimated value is defined by its estimated error due to modeling and input parameter errors. For example, the uncertainty of the effective multiplication factor (k_{eff}) calculated by a reactor core analysis code can be caused by errors of nuclear data, geometrical modeling, material composition data, calculation methodology, etc. The uncertainty of a response can be evaluated by the so-called sandwich rule (Cacuci, 2003) which combines the sensitivity of the response to the input variation with the covariance data between the input parameters. It is well known that the uncertainties of the core physics parameters calculated by a neutronics codes are originated partly from the uncertainty of the evaluated nuclear data library (ENDL) used in the calculation. The MUSAD code evaluates the uncertainties of the core physics parameters such as k_{eff} and few-group cross-sections caused by the uncertainties of the nuclear data library.

In this section, the methodologies adopted in the MUSAD code to evaluate the uncertainties of core physics parameters will be described. The methodology for calculating the sensitivity coefficient and the uncertainty of k_{eff} based on the classical perturbation theory is described in Section 2.1 and the generalized perturbation theory for the sensitivity and the uncertainty of the general responses such as few-group cross sections is presented in Section 2.2. Finally, the uncertainty analysis procedure using the MUSAD code including how to obtain the covariance matrix for aggregated cross-sections is described in Section 2.3.

2.1. Sensitivity and uncertainty of the multiplication factor

In the deterministic approach for an uncertainty analysis, the sensitivity coefficients for the k_{eff} of the system can be derived using the adjoint method based on the classical first-order perturbation theory. First, the forward and adjoint transport equations for the original eigenvalue problem and the forward equations for the perturbed eigenvalue problem are expressed symbolically as follows:

$$(A-\lambda B)\psi = 0, \tag{1}$$

$$(A^*-\lambda B^*)\psi^* = 0, \tag{2}$$

and

$$(A'-\lambda' B')\psi' = 0, \tag{3}$$

where A, B, λ, and ψ are the neutron transport operator except for the fission source term, the fission source operator, eigenvalue of the system, and the angular flux, respectively. Inserting $A' = A+\delta A$, $B' = B+\delta B$, $\lambda' = \lambda+\delta\lambda$, and $\psi' = \psi+\delta\psi$ into Eq. (3) and neglecting the second order terms, we obtain:

$$(A-\lambda B)\delta\psi + (\delta A - \lambda\delta B)\psi - \delta\lambda B\psi = 0. \tag{4}$$

The perturbation of the eigenvalue can be obtained as follow by taking the inner product of Eq. (4) with the adjoint flux as the weighting factor:

$$\frac{\delta\lambda}{\lambda} = \frac{\langle\psi^*(\delta A-\lambda\delta B)\psi\rangle}{\langle\psi^*\lambda B\psi\rangle}. \tag{5}$$

Note that the first term of Eq. (4) was eliminated by using the property of the adjoint operator as follows:

$$\langle\psi^*(A-\lambda B)\delta\psi\rangle = \langle\delta\psi(A^*-\lambda B^*)\psi^*\rangle = 0. \tag{6}$$

Finally, the sensitivity of k_{eff} to the variable X can be expressed as:

$$S_{k_{eff},X} = \frac{\frac{\delta k_{eff}}{k_{eff}}}{\frac{\delta X}{X}} = \frac{\delta k_{eff}}{\delta X}\frac{X}{k_{eff}} = -X\frac{\left\langle\psi^*\left(\frac{\delta A}{\delta X}-\frac{1}{k_{eff}}\frac{\delta B}{\delta X}\right)\psi\right\rangle}{\left\langle\psi^*\frac{1}{k_{eff}}B\psi\right\rangle}. \tag{7}$$

The actual expression of the sensitivity coefficients practically applicable to code implementation can be obtained by substituting the operators with that of the following forward and adjoint multi-group transport equations which were used in the DeCART code:

$$\begin{aligned}&\Omega\cdot\nabla\psi_g(r,\Omega)+\Sigma_{tg}(r)\psi_g(r,\Omega)\\&=\frac{1}{4\pi}\frac{\chi_g(r)}{k_{eff}}\sum_{g'}\nu\Sigma_{fg'}(r)\phi_{g'}(r)+\sum_{g'}\int_{\Omega'}d\Omega'\Sigma_{sg'g}(r,\Omega'\to\Omega)\psi_{g'}(r,\Omega')\\&\quad(g=1,2,\ldots G),\end{aligned} \tag{8a}$$

$$\begin{aligned}&-\Omega\cdot\nabla\psi^*_g(r,\Omega)+\Sigma_{tg}(r)\psi^*_g(r,\Omega)\\&=\frac{1}{4\pi}\frac{\nu\Sigma_{fg}(r)}{k_{eff}}\sum_{g'}\chi_{g'}(r)\phi^*_{g'}(r)+\sum_{g'}\int_{\Omega'}d\Omega'\Sigma_{sgg'}(r,\Omega\to\Omega')\psi^*_{g'}(r,\Omega')\\&\quad(g=1,2,\ldots G),\end{aligned} \tag{8b}$$

where ψ_g, ϕ_g, ψ^*_g, and ϕ^*_g are the forward angular flux, the forward scalar flux, the adjoint angular flux, and the adjoint scalar flux of group g, respectively. The method of characteristics (MOC) is adopted for solving the multi-group transport equations, Eq. (8), in the DeCART code. A backward ray tracing is used in solving the adjoint equation, Eq. (8b) (Cho et al., 2013), while a forward ray tracing is used in solving the forward equation, Eq. (8a).

The solutions of the forward and adjoint transport equations and the scattering cross-section can be expanded with the spherical harmonics and Legendre polynomial as follows:

$$\psi_g(r,\Omega) = \sum_l\sum_m\frac{2l+1}{4\pi}\psi^m_{gl}(r)Y^m_l(\Omega), \tag{9a}$$

$$\psi^*_g(r,\Omega) = \sum_l\sum_m\frac{2l+1}{4\pi}\psi^{*m}_{gl}(r)\bar{Y}^m_l(\Omega), \tag{9b}$$

Download English Version:

https://daneshyari.com/en/article/8068352

Download Persian Version:

https://daneshyari.com/article/8068352

Daneshyari.com