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Systematic and quantitative uncertainty analysis for rod ejection accident of pressurized water reactor

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Abstract

A systematic uncertainty analysis strategy based on stochastic sampling method is developed for best estimate analysis of the rod ejection accident in pressurized water reactor (PWR). Self-developed SAMP module and DAKOTA code are used to randomly sample the multigroup nuclear data, important neutronics and thermal-hydraulic (T-H) input parameters respectively. The 3D coupled reactor core model for Almaraz PWR plant is established using RELAP5/PARCS code system and the uncertainties of the core power, fuel enthalpy and local power distribution are obtained. The results indicate that the uncertainties of nuclear data and input parameters can result in large uncertainty on the results which should be carefully considered.

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Keywords: uncertaity analysis; rod ejection accident; stochastic sampling; nuclear data; DAKOTA

1. Introduction

The rod ejection accident (REA) is most limiting among the reactivity-initiated accidents and is one of the licensing basis accidents for pressurized water reactors (PWR). Due to the pellet cladding mechanical interaction (PCMI) at high burnup which decreases the safety limit for fuel rod failure, conventional 1D analysis method provides insufficient margin and make the results of REA analysis more limiting [1]. Therefore three-dimensional coupled thermal-hydraulics (T-H) and neutronics core model has been used for REA analysis to simulate the strong asymmetric and coupled effects during the accident [2] more precisely.

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Currently the industry such as Westinghouse Company has applied to Nuclear Regulatory Commission (NRC) for using the conservative assumptions in coupled REA analysis to meet the regulation requirements of RG1.77 [3]. The Best Estimate Plus Uncertainty (BEPU) methodology [4, 5] which uses realistic conditions has been approved by NRC for LOCA analysis, and is now actively studied for Non-LOCA accidents mostly using coupled model. A new uncertainty analysis procedure based on stochastic sampling is developed and performed for the BEPU analysis of REA in this paper. Taking the reactor core of Almaraz PWR as the object, the influence of uncertainties of important input parameters on the accident outputs is analyzed systematically and quantitatively.

2. Methodology

2.1. Analysis procedure

The uncertainty analysis procedure developed for PWR REA analysis is demonstrated in Fig. 1. The uncertainties of multigroup nuclear data, fuel rod fabrication tolerance, thermal-hydraulic input parameters and boundary conditions are considered, and they are randomly sampled by self-developed SAMP module and DAKOTA code to generate perturbed nuclear data libraries and input files for lattice and core calculations. The coupled core calculation is performed by RELAP5/PARCS code system under the Parallel Virtual Machine (PVM) environment. The two-group homogenized nuclear data used by PARCS is generated by the lattice code SCALE/TRITON (NEWT). Several Python scripts are developed for linking different calculation steps, generating the input files, extracting the results and performing uncertainty analysis.

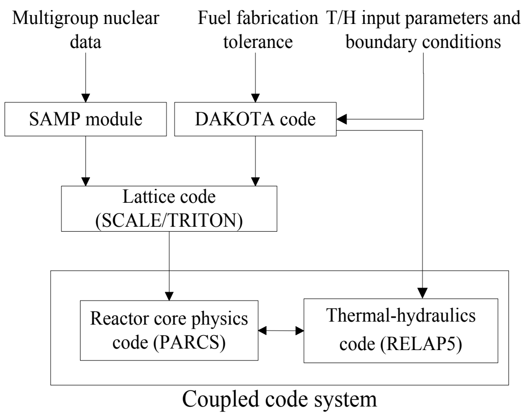

Fig. 1. Flowchart of uncertainty analysis procedure for REA analysis.

According to the Wilks' formula [6], the sample size N of 100 [7,8] in this work is sufficient to guarantee double tolerance bounds defining a tolerance interval for the output variables, which can cover 95% of the variables' probability content with 95% of statistical confidence level. The tolerance interval and standard deviation (std.) are used to quantitatively represent the uncertainties of transient core output variables and radial power distribution. If the probability density functions of output variables are unknown, the maximum and minimum values of the N simulation results of certain output variable can be set as its upper and lower bounds of the tolerance interval respectively [9].

Compared to the two-step uncertainty analysis method [10] developed for reactor core physics analysis, the sampling method can take nearly all input parameters into account, and it make the uncertainty analysis of burnup and branch calculations straightforward. But the N times lattice calculations are more time consuming.