Recent Activities relating on Nuclear Data

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1. Introduction

In the actual reactor power plants, core reactivity prediction is most important from the viewpoint of economics as well as safety. Core design codes together with nuclear data, therefore, are required to give reliable prediction of core reactivity during the plant operation. In this report, the performance of most recent nuclear data in predicting core reactivity is tested against experiments using a continuous energy Monte Carlo code MCNP, and then the method to generate multi-group cross-sections for core design is studied using a nuclear data processing code NJOY.

2. Benchmark Calculations and Results

Objectives of this study are to evaluate prediction accuracy of core reactivity by the use of ORNL ^{235}U, ^{238}U nuclear data and to try to find out the cause of the reactivity difference by comparing cross-sections of ORNL data with JENDL-3.3 or ENDF/B-VI R8.

2.1 Critical Experiments and Calculation Modeling

Benchmark problems are picked up from the International handbook of Evaluated Criticality Safety Benchmark Experiments of NEANSC and a set of CEA Vaduck, DIMPLE, RRC Kurchatov, TCA, B&W, Kritz, MATR is used in this study.

The features of this series of benchmark calculations are that all cases are analyzed by 3-D geometrical model and that some cases cover wide range of temperature from cold (20 deg) to high temperature (250 deg), which is close to the no-load hot conditions of PWR's.

In the calculations, MCNP4C3 was used together with JENDL-3.3 based Library generated byNJOY-99.67 to which a patch for JENDL-3.3 was applied.

2.2Benchmark Resultsl

The calculation results are shown from Fig.1 to Fig.3. We focus the reactivity difference between MCNP and experiment.

Figure 1 shows the dependency of the reactivity difference on fuel enrichment when JENDL-3.3 is used. JENDL-3.3 shows large k_{eff} difference for experiments with low enrichment. The dependency of k_{eff} on ^{235}U enrichment is greatly improved when ORNL ^{235}U and ^{238}U data are used as shown in Fig.2.

Figure 3 shows the dependency of the reactivity difference on H/U in the case that JENDL-3.3 and new ^{235}U, ^{238}U data are used. The prediction of k_{eff} is also improved by 0.25% for the lattices with H/U of 7.2 or lower. However, k_{eff} is still lower for the lattice with H/U of 2.1. Therefore, it is recommended that analyses should be extended to tight lattice experiments with H/U less than 2.1 (TCA, TRX, PSI).

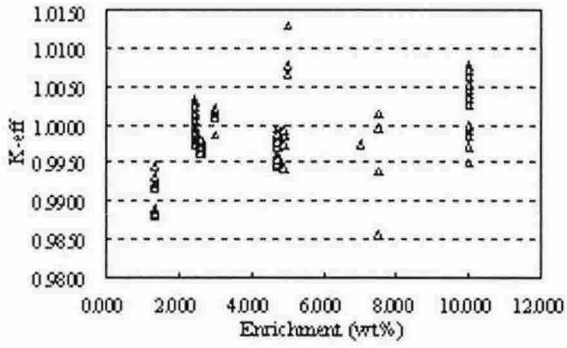

Figure 1. keff vs Enrichment by MCNP4C3(Preliminary) [JENDL-3.3]

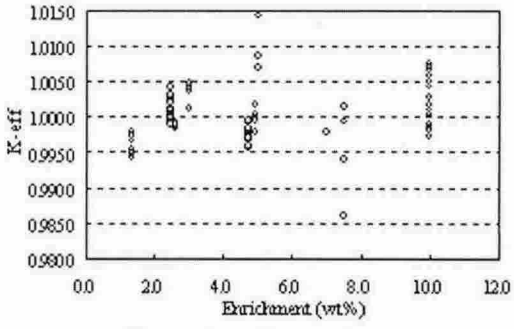

Figure 2. keff vs Enrichment by CNP4C3(Preliminary) [JENDL-3.3+ORNL(^{235}U,^{238}U)+ENDF/B-VI r8(O,H)]

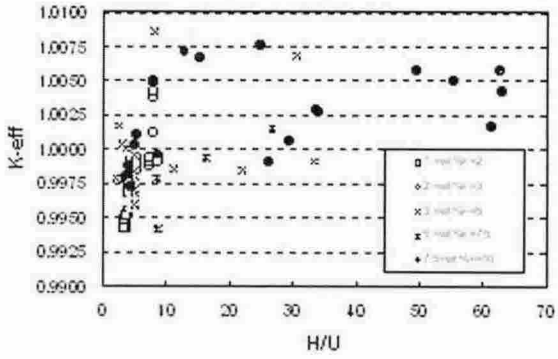

Figure 3. keff vs H/U by MCNP4C3(Preliminary) [JENDL-3.3+ORNL(^{235}U,^{238}U)+ENDF/B-VI r8(O,H)]

3. Multi-group Cross-section Library

3.1How to generate multi-group Cross-sections

The nuclear data processing code NJOY is widely used for generating multi-group cross-sections. However, it is not necessary clear how to generate the best multi-group cross-sections to be applied to PWR core design.

In this study, we investigated the dependency of the reactivity difference between a core design code with multi-group cross-sections and a continuous energy Monte Carlo code to know how many energy groups is needed and what is the best weighting spectrum. The 70- and 187-energy group structures were tested mainly. PHOENIX-P was used as a design code, which is deterministic and employs the combined method of Pin Cell Coupling and S4 calculation.For PHOENIX-P, its cross-section library was made based on JENDL-3.3 nuclear data using NJOY. As a statistical code, a Monte Carlo code MVP was used, which had been developed by JAERI and for which JENDL-3.3 based cross-section library was supplied by JAERI.

In the calculation of reactivity difference, we used a cell model with a pitch of 1.3133cm to preserve the moderator to fuel volume ratio of a PWR 17×17 standard fuel assembly with cell pitch of 1.26 cm.

3.2 Weighting Spectrum

We checked the following three weighting spectrum:

(1) Maxwell+1/E+χ[IWT=4]
(2) EPRI-CELL Spectrum [IWT=5]
(3) EPRI-CELL Spectrum+ Flux Calculator [IWT=-5]

In the case of IWT=-5, heterogeneous and homogeneous systems were tested by changing the input of NJOY.

For heterogeneous cases, a heterogeneous factor, denoted as beta, was treated in two ways; one is to keep it constant and the other is to change the value according to moderator to fuel volume ratio. The beta value must between 0.0 and 1.0 from its physical meaning.

3.3 Dependency of k_{inf} on Enrichment or H/U

The obtained results regarding the dependency of reactivity difference with respect to enrichment or moderator to fuel volume ratio V_m/V_f are shown in Fig.4 and Fig.5.

The use of 70G-analytic spectra (IWT=4,5) shows strong enrichment dependencies. On the contrary, the 187G-analytic spectrum does not show any enrichment dependency. Flux calculator spectrum improves enrichment dependency in 70G (IWT=-5), but reactivity difference moves to minus and k-bias seems to appear. The reactivity difference in 187G also becomes negative when flux calculator is used (IWT=-5). Flux calculator seems to work well to avoid enrichment dependency, but cause reactivity bias. Check is needed using other code.

From Fig.5, we can see analytical spectra do not show any dependency on V_m/V_f. However, when flux calculator is used, a large dependency appears. The issue might be a problem on reactor physics.

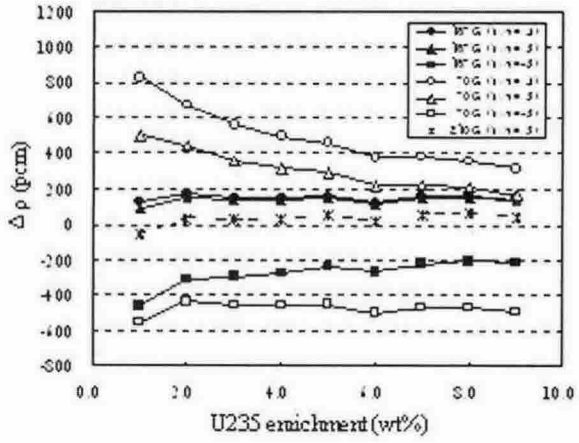

Figure 4. Dependence of reactivity difference between PHOENIX-P and MVP on ^{235}U enrichment (JENDL-3.3)

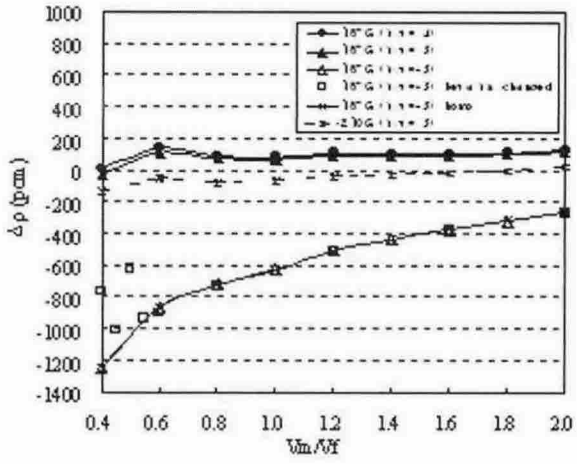

Figure 5. Dependence of reactivity difference between PHOENIX-P and MVP on V_m/V_f(JENDL-3.3)

4. Conclusions

The reactivity prediction against critical experiments is greatly improved by employing ^{235}U, ^{238}U nuclear data reevaluated by ORNL.

The ^{235}U enrichment dependency will be solved by employing more energy meshes or by using flux calculator. However, when flux calculator is used, k_{inf} bias seems to appear. The more direct comparison using MCNP with cross-sections generated by NJOY is needed.

From the results of V_m/V_f dependency of k_{inf}, the calculation method of effective resonance cross-section might still have problems. More study is needed on this point.