<u>Nuclear data for Non-refueling core design</u>

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For design of innovative control system and safety characteristic of the Non-refueling core design of long life, a series of critical experiments is conducted at the fast critical facility, FCA of JAEA-Tokai. To quantitatively estimate the uncertainty reduction through critical experiments, an uncertainty reduction ratio (UR) is introduced, using the cross section error. Additionally with sensitivity analysis of the cross sections, important cross sections are clarified for burn-up calculation.

1. Introduction

The 4S, Super Safe, Small and Simple~~,~~ reactor (Fig.1) is a kind of fast reactor core in which burn-up reactivity loss is compensated by decrement of neutron leakage probability with movement of reflector. For extending core life of the Non-refueling core up to 30 years generating 30MW thermal power, a core of 2.5m height has been designed and studied in which 2~~0~~<u>1</u>%/24%-Pu-enriched Pu-U-Zr metallic fuel pins are loaded (Fig.2) [1,2].

Present document is based on extractions from papers of authors [3,4].

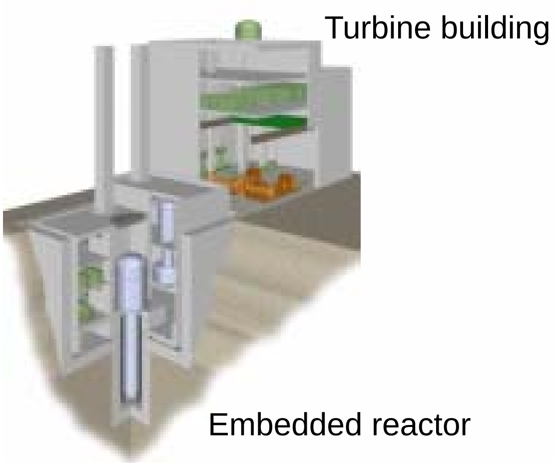

Fig. 1 Overview of 4S Reactor

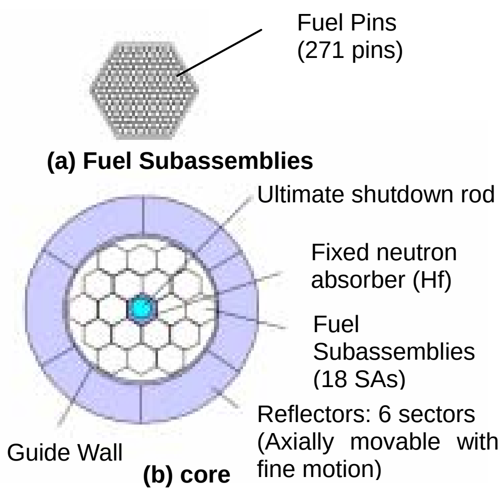

Fig. 2 Design of the Non-refueling core

2. Critical experiments [3].

For design of innovative control system and safety characteristic of the Non-refueling core of long life, we have to verify and improve neutronics calculation methods. However, there are few experimental data ~~measured focusing~~focused on reflector reactivity, small (zero or negative) Na void reactivity~~,~~ etc.. For the verification of the design methods, ~~A~~ a series of critical experiments is conducted at the fast critical facility, FCA of JAEA-Tokai. A core of metallic fuels of Pu and Pu+U surrounded by massive reflector of stainless steal has been mocked up and measurements of several kinds of reactivity and reaction rate distribution ~~has~~ have been conducted. The measured data have been analyzed by conventional deterministic diffusion~~-~~/~~-~~transport codes and continuous energy Monte Carlo codes. By the comparison of calculated one to the data, prediction accuracies of neutronics codes have been clarified.

The experimental core (FCA XXIII core) was constructed to focus on the behavior of neutron leakage in radial direction and the neutron spectrum. As shown in Fig. 3, the core has a core region (about 30 cm in radius and 111 cm in height) with a sodium (Na) channel (about 9 cm in radius) along the center axis. The core region is divided into two zones, the inner zone and the outer zone. The inner zone is composed of Pu fuel plates, natural U plates, Zr plates and Na plates to simulate the composition of the Non-refueling core. The composition of the outer zone is slightly different from that of the inner zone; enriched U plates instead of some of Pu plates were used because of the insufficient inventory of fissile materials. The core region is surrounded by the Na region of 5cm thickness and the reflector region of 33cm thickness.

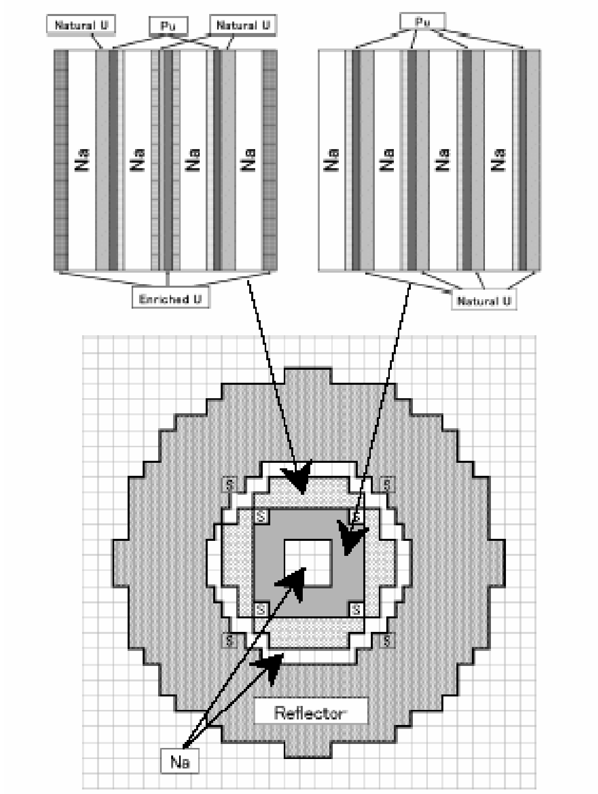

Fig. 3 Cross sectional view of the experimental core (FCA XXIII core)

Measurements have been made for criticality (keff), central reaction rate ratios, reaction rate distributions, Na void reactivity worth and reflector reactivity worth. Especially, to get the leakage and no-leakage information in the Na void reactivity worth, the detailed distribution of the Na void reactivity worth has been measured. The experimental core first went critical in July 2004.

3. Neutronic uncertainty reduction technique [4].

Using the experimental data, the bias factor, the ratio of measured neutronics characteristics to the calculated ones, is obtained. The uncertainty of this bias factor is also calculated. To quantitatively estimate the uncertainty reduction through critical experiments, an uncertainty reduction ratio (UR) is introduced, using the cross section error. By using UR, better experimental core can be mocked up and required accuracy for experiments have been identified to reduce the uncertainty of the bias factor, i.e., to improve the accuracy of design calculation of the target the Non-refueling core. [5,6].

As an example, let us show the uncertainty reduction (UR) for the conversion ratio. The UR is obtained for the inner core (IC) and the outer core (OC) in the Non-refueling core as a function of the experimental error as shown in Fig.4.

The UR is almost 0.85 when experimental error is zero. This means that the uncertainty is reduced by 85% when using the critical experiment. The measurement with more than 5% error is meaningless for the reduction of the uncertainty of the target Non-refueling core. Therefore, accurate measurement is necessary. The large UR is caused by the fact that they have very similar nuclide-wise component of uncertainty (S^tVS) due to the cross section error as shown in Fig. 5. Here S and V denote the sensitivity coefficient and cross section covariance matrices, and superscript t shows the transpose of the matrix. As shown in Fig. 5, ^{238}U and ^{239}Pu are the main component in S^tVS for the case of conversion ratio. The similar trend for these components leads to the satisfactory reduction of uncertainty.

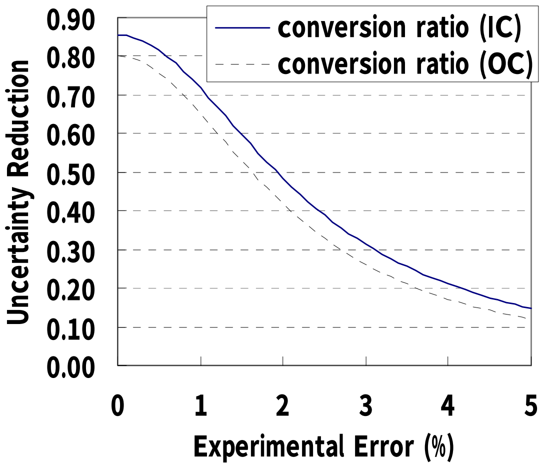

Fig. 4 UR vs. experimental error (conversion ratio)

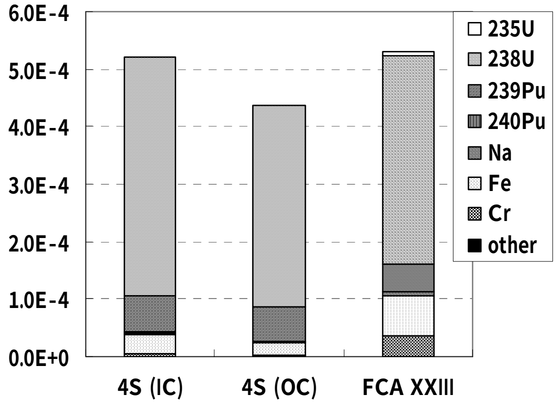

Fig. 5 Nuclise-wise component of SVS (conversion ratio)

4. Sensitivity analysis of burn-up depletion

For achievement of long core life of 30 years without refueling, prediction of burn-up reactivity depletion is important. With sensitivity analysis of the cross sections, important cross sections are clarified for burn-up calculation.

Calculations are carried out with 2-dimensional (R-Z) diffusion code with 70 energy group

JFS-3-J3.3 library (added lumped FP data of JFS-3-J32R library on it). Seven group effective cross-sections were created, and burn-up calculation was performed. Sensitivity coefficients are calculated directly with fluctuation of buru-up depletion reactivity resulting from small change of seven group effective cross-sections.

Sensitivity coefficients of the fission cross section of ^{235}U, ^{239}Pu, and ^{241}Pu (fissile nuclides) are positive (when the cross-section is increased, depletion reactivity is also increased), and a fertile nuclides have negative values. Sensitivity coefficients of the capture cross section of ^{238}U and ^{240}Pu (fertile nuclides) are negative, and the sensitivity coefficient of ^{239}Pu capture cross section is positive. These results are due to accumulation of fissile nuclides. Important nuclides and reactions are fission cross section of ^{239}Pu and capture cross section of ^{238}U (Fig. 6) [7].

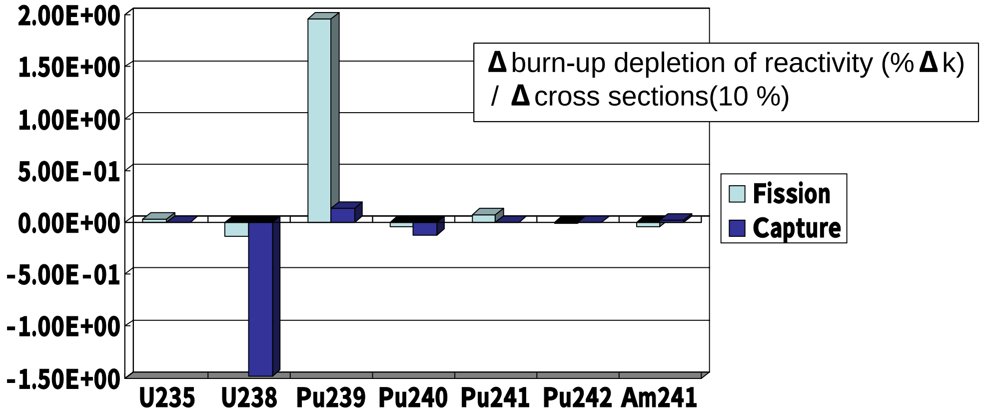

Fig. 6 Sensitivity coefficients of buru-up depletion reactivity

5. Summary

- In the Non-refueling core, burn-up reactivity loss of 30 years is compensated by decrement of neutron leakage probability with movement of reflector. And small (zero or negative) Na void reactivity is important for safety core characteristics of the Non-refueling core.
- For the verification of the design methods, series of critical experiments is conducted at FCA.
- To quantitatively estimate the uncertainty reduction through critical experiments, an uncertainty reduction ratio (UR) is introduced, using the cross section error.
- With sensitivity analysis of the cross sections, important cross sections are clarified for burn-up calculation

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