Benchmark Experiment for Physics Parameters of Nitride Fuel LMFBR at FCA

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ABSTRUCT

Results of the benchmark test of reactor physics parameters of a nitride-, metallic- and oxide-fuel LMFBR was reported. Sodium void reactivity worth and plutonium and B_4C sample worth were measured systematically on the mockup assemblies with different fuel composition. The analyses were made by using the JENDL-3.2 cross-section library and JAERI's standard calculation system for fast reactor neurotics. Predicts of criticality, sample worth and sodium void reactivity worth in the central small zone agreed well with the measured values in the three assemblies. The difference of calculation accuracy among the nitride-, metal- and oxide-fuel assembly was found in the sodium void reactivity worth of the large zone and the discrepancy between the calculated and the measured value was obvious in the nitride-fuel assembly.

1. INTRODUCTION

The benchmark experiment for reactor physics parameters of a nitride fuel LMFBR has been conducted on the Japan Atomic Energy Research Institute's (JAERI's) fast critical assembly (FCA) facility in order to test calculated predictions of physics parameters in the core design. A sodium void reactivity, which is a sensitive physics parameter to the fuel composition, was measured on the mockup core of nitride fuel LMFBR. The plutonium (PU92) and B_4C sample worth were also measured as the standard reactivity effects. Before the nitride fuel core experiment, the mockup experiment for a metallic fuel core has been made as another advanced fuel LMFBR[(1),(2)]. The mockup experiment for an oxide (MOX) fuel core has also been made to get data of conventional fast reactors as reference. The sodium void reactivity worth and the sample worth were measured systematically on the assemblies with different fuel composition using the same sample and the same measurement method.

The experiment analyses of the nitride fuel core were made by using the JENDL-3.2 cross-section library[(3)] and the JAERI's standard calculation system for fast reactor neutronics. The calculation accuracy of sodium void reactivity worth, PU92 sample worth and B_4C sample worth in the nitride fuel core were examined on the comparison with the results in the metallic- and the oxide-fuel assembly.

2. EXPERIMENTS

2.1 ASSEMBLY DESCRIPTION

The mockup experiment of the nitride fuel LMFBR has been made in FCA assembly XX-2. The assembly is a zone type partial mockup core and has the cylindrical test region with the nitride fuel composition at the core center. The test region is 52 cm in diameter and is 91 cm in height. The driver region and a radial blanket (DUB; depleted uranium block) are placed outside of the test region. FCA has a horizontal separate type structure and consists of fixed- and movable-half assembly. A cross sectional view at the mid plane of assembly is shown in Fig.1 and the core configuration of fixed-half assembly in Fig. 2. The metallic-fuel mockup experiment has been performed in FCA assembly

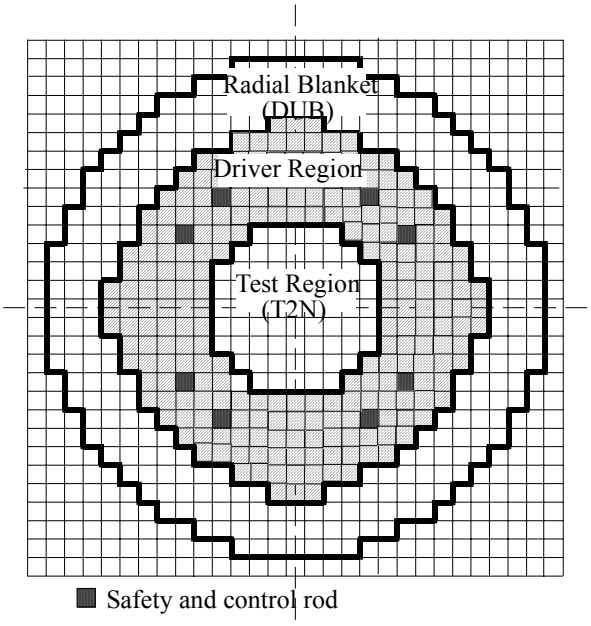

Fig. 1 Cross sectional view of FCA assembly XX-2

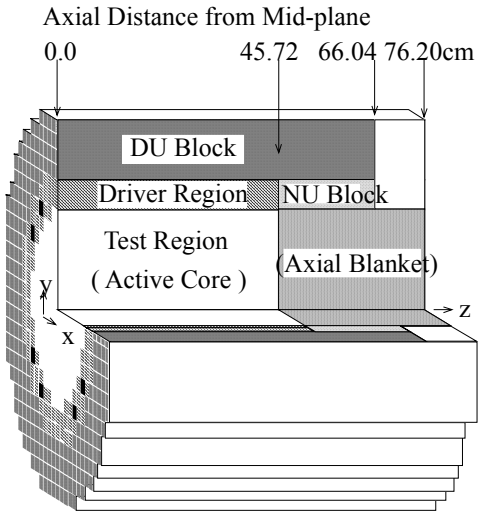

Fig. 2 Configuration of FCA assembly XX-2

XVI-2 and the oxide-fuel mockup experiment in FCA assembly XVII-1. Both assemblies are the zone type partial mockup core and have the central test region with metallic- and oxide-fuel composition respectively. The test region of both assemblies is 69 cm in diameter and is 91 cm in height.

The nitride fuel unit cell, which consists of plutonium, natural uranium, sodium metal plates and aluminum nitride (AlN) ceramic plates, was used in the fuel drawer of test region (T2N drawer). The dimension of unit cell is 5.08 x 5.08 x 5.08 cm. The nine nitride fuel unit cells are inserted in the

active core zone of T2N drawer and the six SB cells with depleted uranium oxide plates and sodium plates are inserted in the axial blanket zone of T2N drawer. The mockup assembly was constructed by loading the T2N and other type drawers into the 5.5 cm square stainless steel matrix tubes. The plate arrangements of unit cell in the nitride-, metallic- and oxide-fuel are shown in Fig. 3. Plutonium enrichment and other core data are given in Table 1.

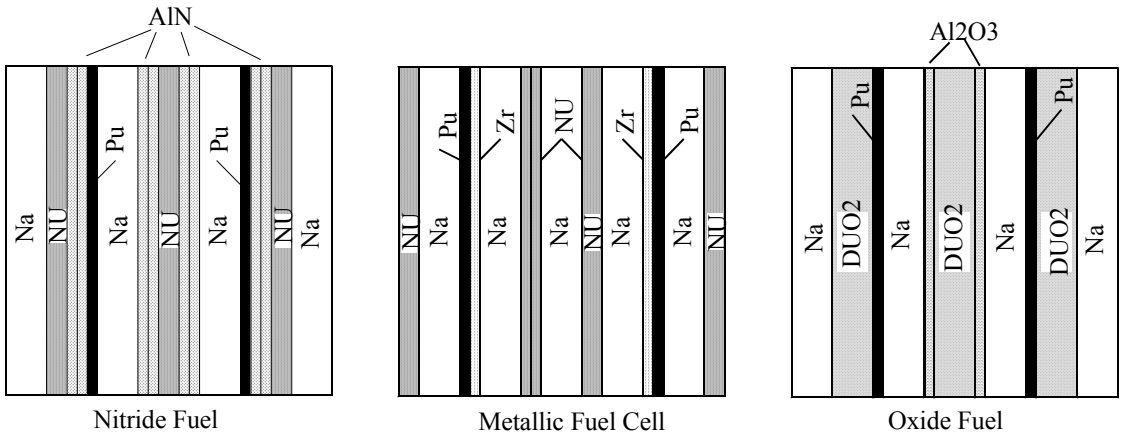

Fig. 3 Plate arrangement in the unit cell of nitride-, metallic- and oxide-fuel assembly

Table 1 Core data of nitride-, metal- and oxide-fuel core

Assembly Name	Nitride-Fuel Core FCA XX-2	Metallic-Fuel Core FCA XVI-2	Oxide-Fuel Core FCA XVII-1
Test Region			
Diameter (cm)	52	69	69
Fuel	Pu + NU	Pu + NU	Pu + DUO_2
Fissile Enrichment (%)	12	11	14
Driver Region			
Diameter (cm)	109	93	95
Fuel	Pu + EU	Pu + EU	
Fissile Enrichment (%)[(a)]	20	23	34

(a) ($^{239}Pu + ^{241}Pu + ^{235}U$) / HM

2.2 MEASURMENTS

The sodium void reactivity worth was measured in the central 3 x 3 drawers of the fixed-half

assembly. The voided zone in the 3 x 3 drawers is illustrated in Fig. 4. The zone void reactivity worth was measured on the (1Z to 2Z) zone, (1Z to 6Z) zone and (1Z to 9Z) zone in the central drawers. In the measurement of central (1Z to 2Z) zone void reactivity worth, sodium plates in 18 cells were replaced to stainless steel void cans at the axial position 1Z and 2Z in 3 x 3 drawers. The reactivity worth was measured by the deference of the control rod positions at the critical states before and after the replacement of sodium plates. The control rod is calibrated on the reactivity scale measured by the positive period method. The reactivity scale is converted to the absolute unit in Δk/k using the delayed neutron data given by Tomlinson and Saphier. The experimental error is considered for the accuracy of the control rod position and of the core temperature correction. In addition to the zone void reactivity worth, a step void reactivity worth was measured at each axial position from 1Z to 9Z in the 3 x 3 drawers.

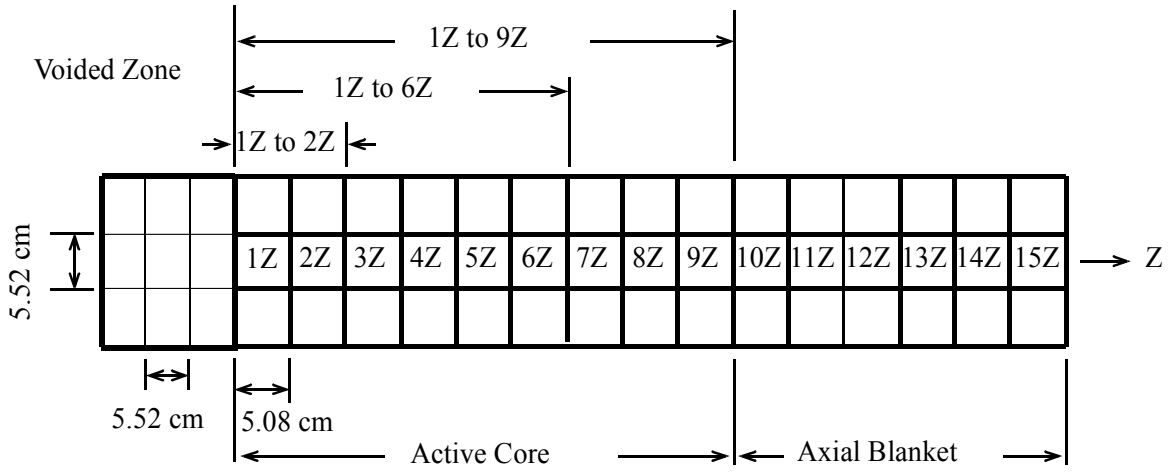

Fig. 4 Central 3 x 3 drawers in the fixed-half assembly on sodium void worth measurement

The reactivity worth for the PU92 sample (plutonium weight; 70.23g), B_4C (20%) sample (^{10}B weight; 4.22 g) and B_4C (90%) sample (^{10}B weight; 18.96 g) was measured at the 1Z cell in the center drawer. The same samples were used in the nitride, metallic and oxide fuel assembly. The calibrated control rod is used in the sample worth measurements. The measured values of sodium void reactivity worth and sample worth in the three assemblies are given in Table 2. The sodium void reactivity worth, which is sensitive to the fuel composition, shows a large difference among the three assemblies, while the difference among the assemblies is ~12% on the PU92 sample worth measured as a standard reactivity.

Table 2 Measured reactivity worth in the nitride metal and oxide fuel core

(unit: 10^{-4} Δk/k)

	Distance[(a)] cm	Nitride Core	Metal Core	Oxide Core
Sodium void worth				
1Z to 2Z	0.0 – 10.16	0.695 ±0.032	1.922 ±0.026	1.006 ±0.027
1Z to 6Z	0.0 – 30.48	1.079 ±0.032	4.173 ±0.026	1.881 ±0.027
1Z to 9Z	0.0 – 45.72	0.079 ±0.032	3.741 ±0.026	0.968 ±0.027
Sample worth				
PU92	2.54	1.785 ±0.032	1.770 ±0.024	1.983 ±0.030
B_4C (20%)	2.54	-1.211 ±0.032	-1.482 ±0.024	-2.288 ±0.030
B_4C (90%)	2.54	-3.955 ±0.032	-4.752 ±0.024	-7.125 ±0.030

(a) from the mid plane

In applying the perturbation theory based on the diffusion theory, reactivity worth is given by

$$\frac{1}{k_{eff}} - \frac{1}{k^*_{eff}} = \frac{1}{\sum_g \nu\Sigma_{f,g}\phi^*_g \cdot \sum_{g'} \chi_{g'}\psi_{g'}}$$

$$\times\left[-\sum_g \delta D_g \nabla\psi_g \bullet \nabla\phi^*_g - \sum_g \delta\Sigma_{a,g}\phi^*_g\psi_g - \sum_g \phi^*_g \sum_{g'} \delta\Sigma_{s,g'\to g}\left(\psi_g - \psi_{g'}\right) + \frac{1}{k^*_{eff}} \sum_g \chi_g\psi_g \sum_{g'} \delta\nu\Sigma_{f,g'}\phi^*_{g'} \right]$$

where Φ_g : neutron flux in energy group g

Ψ_g : adjoint neutron flux

Parameters marked* are those in the perturbed core. The first member in the bracket is a leakage term and the second is an absorption term. The third member including a subtraction of adjoint neutron flux is a scattering term and the fourth is a source term. The sodium void reactivity worth consists of a positive reactivity component and a negative reactivity component. The scattering term is a main factor in the positive reactivity component and the leakage term is a main factor in the negative reactivity component. We can find the combination of positive and negative reactivity component in the axial distribution of the step void reactivity worth shown in Fig. 5. The large positive reactivity worth at the core center decreases toward the axial core boundary by increasing the leakage term.

The main factor in the PU92 and B_4C sample reactivity worth is the source term and the absorption term.

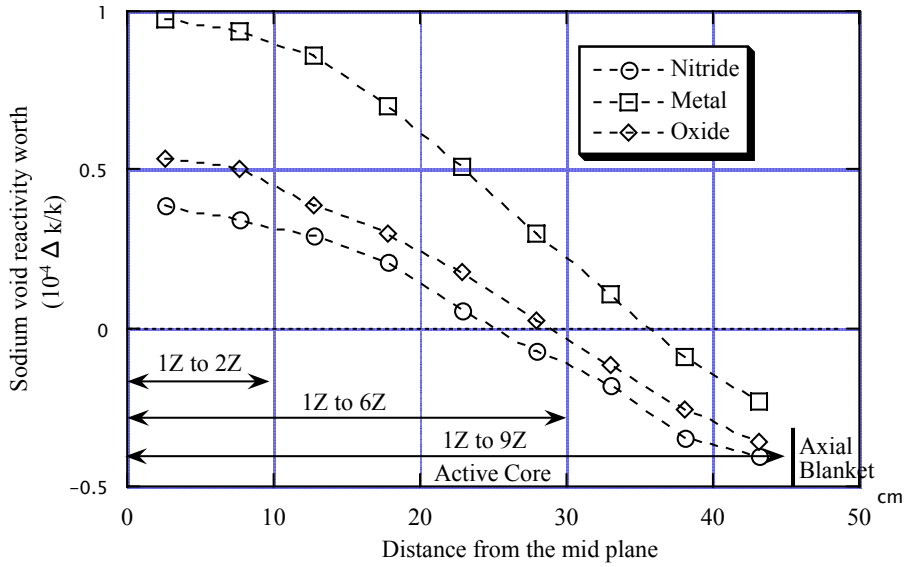

Fig. 5 Axial distribution of step void worth in nitride-, metal- and oxide-fuel assembly

3. ANALYSIS

3.1 CALCULATION METHOD

The JFS-3-J3.2[4] group constant set with a 70-energy group structure (0.25 lethargy width) has been generated from the JENDL-3.2 cross section library. Infinite-medium cell calculations, which involve two steps- resonance shielding for heavy isotopes by the "table lookup" method and calculation of the flux fine structure by the collision probability method, are made and cell averaged group constants are prepared by flux- volume weighting to preserve the reaction rates in the heterogeneous cell. Anisotropic diffusion coefficients are also prepared by Benoist's formula[5]. These calculations are made using the SLAROM code[6]. The same data and method was used in the

analyses of the nitride-, metallic- and oxide-fuel assembly.

3.2 CRITICALITY

The diffusion calculation using 70-energy group constant including anisotropic diffusion coefficients was made in x-y-z geometry modeling the reference core configuration (shown in Fig. 1 and 2). The core calculation is made using the POPLARS code[7]. Transport and mesh-size corrections were applied to the k_{eff}. The transport correction is taken from transport and diffusion calculations in r-z geometry. The Sn calculation is made by the TWOTRAN-II code[8]; order of n is eight and a correction factor ($1-\mu$) is applied to elastic scattering cross section.

The calculated-to-experiment (C/E) value for k_{eff} is given in Table 3. The discrepancy between the measured and the calculated value is less than 1%.

In addition to these calculations, the continuous energy Monte Carlo calculation by the MVP code[9] was made using the whole core model of the assembly. The difference of k_{eff} between the traditional deterministic method and the Monte Carlo calculation is small and is 0.1 % to 0.6 %.

Table 3 Results of k_{eff} in nitride-, metal- and oxide-fuelled assemblies

	Nitride Core	Metal Core	Oxide Core
Experiment	1.00238±0.00005	1.00354±0.00013	1.00743±0.00013
C/E	0.998	0.990	0.994

3.3 SODIUM VOID REACTIVITY WORTH

The zone void reactivity worth measured on three steps was calculated by the exact perturbation method based on diffusion theory using the x-y-z geometry and the 70-eneregy group constants. The calculation is made using the PERKY code[10]. The transport corrections were applied to the non-leakage term (summation of absorption, scattering and source term) and the leakage term independently. The results are shown in Table 4. Since the sodium void reactivity worth consists of positive non-leakage term and negative leakage term, a ratio of

Table 4 C/E values of sodium void reactivity worth

	Nitride Core	Metal Core	Oxide Core
1Z to 2Z	1.071	1.014	0.996
1Z to 6Z	1.168	1.010	0.958
1Z to 9Z	2.093	1.018	0.884

leakage term to non-leakage term is an important parameter on the examination of calculation accuracy.

The non-leakage term is a dominant component on the central (1Z to 2Z) zone void reactivity worth and the ratio of leakage term (absolute value) to non-leakage term is 0.02 to 0.05 in the nitride-, metallic- and oxide-fuel assembly. Predictions of the central zone void reactivity worth agree well with the measured values in the three assemblies.

In the large (1Z to 9Z) zone void reactivity worth, the ratio of leakage term to non-leakage term is 0.97 in the nitride-fuel assembly and is 0.41in the metallic-fuel assembly. The ratio of the oxide-fuel assembly is in the middle. Predictions become various results in the three assemblies according to the ratio of leakage term to non-leakage term. The discrepancy between the calculated and the measured reactivity worth is obvious in the nitride-fuel assembly in which the absolute value of leakage term closes to the non-leakage term.

3.4 SAMPLE REACTIVITY WORTH

The first-order perturbation calculation based on the diffusion theory in 70-energy groups was made on the evaluation of the PU92, B_4C (20%), and B_4C (90%) sample reactivity worth in the r-z geometry. The C/E values are shown in Table 5. The calculated values for the sample worth show a good agreement with the measured results in the range of – 4 % to + 15 %.

Table 5 C/E values of sample worth

	Nitride Core	Metal Core	Oxide Core
PU92	1.033	0.993	1.013
B_4C (20%)	1.143	0.967	1.000
B_4C (90%)	1.132	1.055	1.120

CONCLUSIONS

The benchmark experiments to test the calculation accuracy of the sodium void reactivity worth and the sample worth were made in the nitride-, metallic- and oxide-fuel assembly. The experiments were analyzed by using the JENDL-3.2 cross-section library and JAERI's standard calculation system for fast reactor neutronics. The calculation predicts well the physic parameters for k_{eff} and the sample worth of PU92, B_4C (20%) and B_4C (90%).

In the sodium void reactivity worth, the calculation predicts well the central zone void reactivity worth in the nitride-, metallic- and oxide-fuel assembly. The prediction accuracy of large zone void reactivity worth depends on the ratio of leakage to non-leakage term and discrepancy between the calculated and measured reactivity worth become large in the nitride-fuel assembly.

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