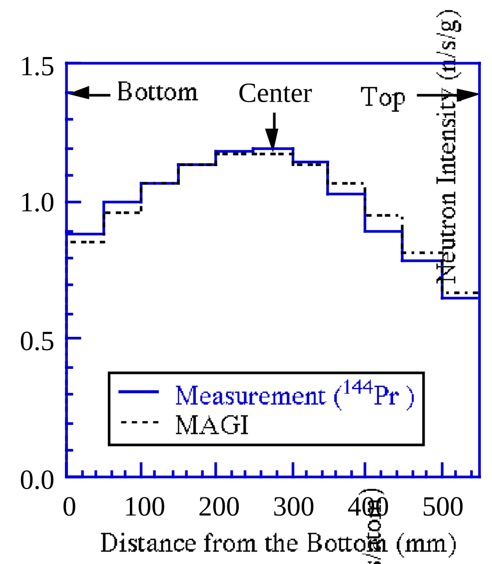
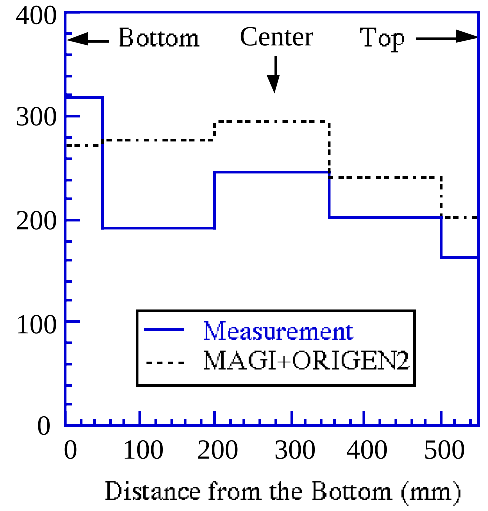


Fig. 3 Axial Burnup Distribution



xial Neutron Intensity Distribution Fig. 4

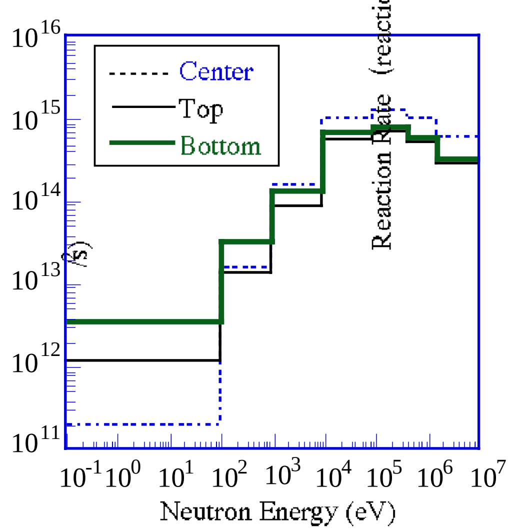


Fig. 5 Comparison of Neutron Spectrum

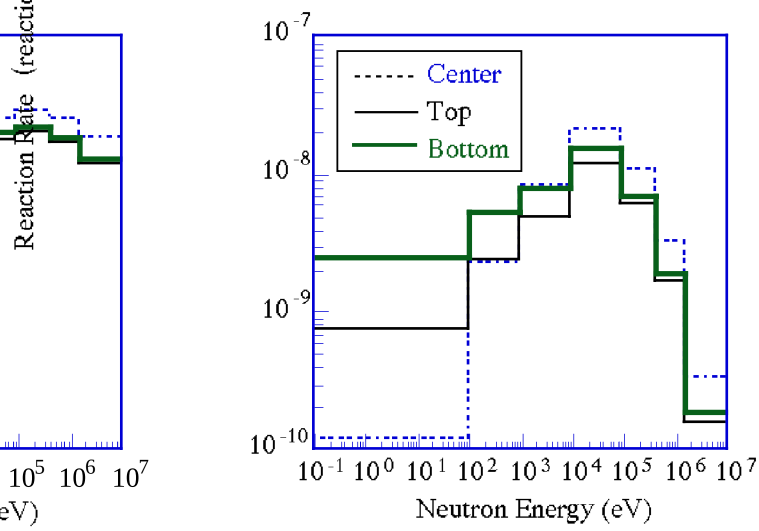


Fig. 6 Comparison of ^{243}Am(n, γ) Reaction Rate

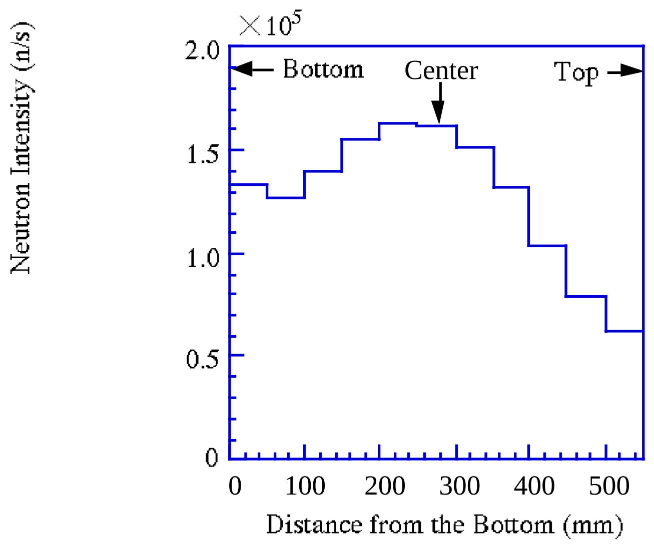


Fig. 7 Axial Neutron Intensity Distribution from ^{244}Cm

Table 5. Neutron Intensity of Individual Nuclide (by MAGI Calculation)

Nuclide	Neutron Intensity (n/s/Subassembly)			Ratio [illegible]
	(α , n) Reaction	Spontaneous Fission	Total	
^{238}Pu	2.8×10^{5}	4.5×10^{4}	3.2×10^{5}	11.0
^{239}Pu	8.3×10^{4}	—	8.3×10^{4}	2.85
^{240}Pu	1.2×10^{5}	6.1×10^{5}	7.3×10^{5}	25.1
^{242}Pu	—	1.3×10^{5}	1.3×10^{5}	4.62
^{241}Am	2.0×10^{5}	—	2.0×10^{5}	6.81
^{242}Cm	4.8×10^{3}	2.3×10^{4}	2.8×10^{4}	0.96
^{244}Cm	1.2×10^{4}	1.4×10^{6}	1.4×10^{6}	[illegible]
Total	6.9×10^{5}	2.2×10^{6}	2.9×10^{6}	[illegible]

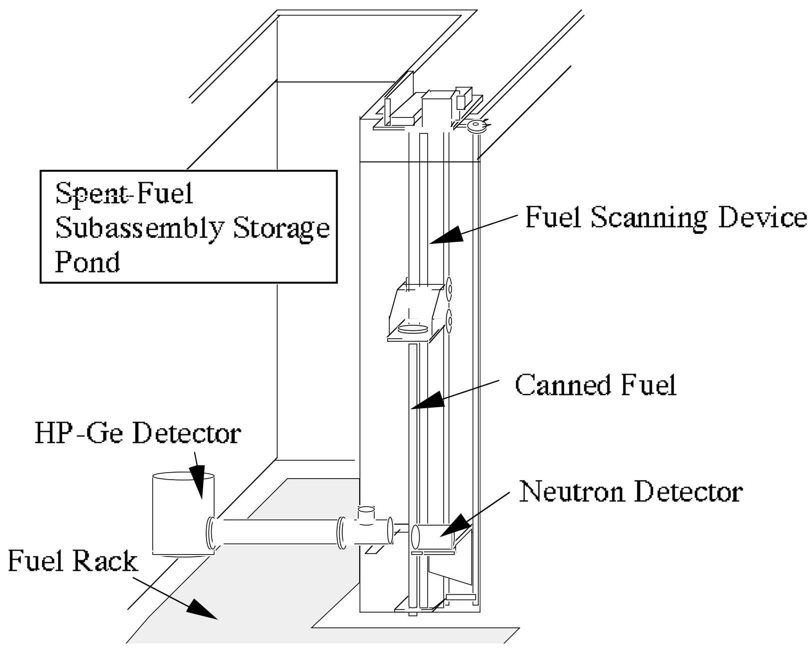


Fig. 1 Schematic Diagram of Measurement System

$$(Ri)=(Mij)(Sj) \rightarrow (Sj)=(Mij)^{-1}(Ri)$$

Ri : Neutron Counting Rate (cps)
Rj : Neutron Intensity (n/s)
[illegible] : Response Matrix

Fig. 2 Neutron Detector Response Matrix

Table 1 Outline of Measured Spent Fuel Subassembly

Fuel Composition	^{235}U Enrichment : 12.8 % Pu Content : 26.8 %
Irradiation Condition	Core Resident Period : 1989.5.8～1992.6.17 1) ϕ_{total} : 3.4×10^{15} $n/cm^2/s$ Irradiation Days : 181 days (at the Second Row) 2) ϕ_{total} : 2.6×10^{15} $n/cm^2/s$ Irradiation Days : [illegible] days (at the Fourth Row)
Φ_{total}	1.07×10^{23} n/cm^2 (Subassembly Averaged)
Burnup	62,500MWd/t (Subassembly Averaged)
Cooling Time	1,900 days (Measured Date : 1997.8.30)

Table 2 Condition of Detector Response Calculation

Calculation Code	MCNP-4A
Cross Section	FSXLIB(JENDL-3.2)
Neutron Source Spectrum	Spontaneous Fission Watt Type (^{244}Cm) (α、 n) Reaction α-ray Energy : 5.5MeV
History	5,000,000
FSD	Less than 1% for Detector Responses

Table 3 Condition of Neutron Flux and Burnup Calculation

Item	MAGI
Cross Section	JFS-3-J2(JENDL-2)
Geometry	3D Hex-Z
Energy Group	7
Flux Calculation	Diffusion
Burnup Calculation	Matrix Exponential

Table 4 Comparison of Neutron Intensity

Neutron Intensity (n/s/Subassembly)		C/E
Measurement	MAGI + ORIGEN2	
2.39×10^6	2.91×10^6	1.22

"MAGI" calculation shown in Table 5 indicates that neutron intensity from ^{244}Cm amounts to about 50% of the total number. Therefore, the neutron intensity depends on the ^{244}Cm distribution in the fuel which had been mainly produced by ^{243}Am (n, γ) reaction.

In order to investigate the ^{244}Cm distribution, the neutron spectra and ^{243}Am (n, γ) reaction rates at the top, center and bottom of the fuel region are compared in Fig. 5 and Fig. 6, respectively. As the JOYO Mk-II driver fuel has axial reflectors of stainless steel with an insulator pellet of depleted uranium between MOX fuel and the axial reflector, the neutron spectrum at the bottom of the fuel region is softer than that of the center. This effect is less at the top because of the neutron absorption by the control rod. As a result, ^{243}Am (n, γ) reaction rate at the bottom is higher than that of the top, and the increase of the neutron intensity from ^{244}Cm as shown in Fig.7 was also due to the same reason. Similar phenomena was observed concerning the ^{243}Am distribution from Am isotope ratio analysis conducted in post irradiation examination.[6] However, the increase of the measured neutron intensity at the bottom end was significant, and is even larger than that of the center. It appears due to the error in solving the inverse matrix of detector response, so further investigation is required.

5. Conclusion

Neutron intensity of spent JOYO Mk-Ⅱ fuel with a burnup of 62,500MWd/t and cooling time of 5.2 years was measured and compared with the calculation based on the JOYO core management code system. The measured neutron intensity in the whole fuel was about 2.4×10^{6}n/s, and the average C/E approximately 1.2 was obtained. It was found that the axial neutron intensity didn't simply follow the burnup distribution, and the neutron intensity was locally increased at the bottom end of the fuel region due to an accumulation of ^{244}Cm.

Acknowledgements

The authors would like to note the contribution of Mr. T. MASUI and Mr. T. SAIKAWA of Inspection Development Company for the neutron and gamma-ray measurement, and Mr. M. NAGATANI of Information Technologies Japan Inc. for the neutron flux and burnup calculation by "MAGI". We also appreciated the cooperation of Mr. O. SATO and Ms. S. FURIHATA of Mitsubishi Research Institute Inc. for the detector response calculation.

References

1) J. F. Brismeister Ed.: "MCNP-A General Monte Carlo N-Particle Transport Code Version 4A," *LA*- 12625-*M* (1993)
2) K. KOSAKO et al.: "FSXLIB-J3: MCNP Continuous Energy Cross Section Library Base on JENDL-3," *JAERI-M* 91-187 (1991)
3) T. NAKAGAWA et al.: "Japanese Evaluated Nuclear Data Library Version 3 Revision-2: JENDL-3.2," *J. Nucl. Sci. Technol.*, **32**, 1259 (1995)
4) T. AOYAMA et al.: "Core Management Experience of the Experimental Fast Reactor JOYO," *Proc. Int. Conf. Fast Reactor Core and Fuel Structural Behavior, British Nuclear Energy Society, UK*, p.299 (1990)
5) H. TAKANO and K. KANEKO: "Revision of Fast Reactor Group Constant Set JFS-3-J2," *JAERI-M* 89-141 (1989)
6) S. KOYAMA et al.: "Results of Am Isotopic Ratio Analysis in Irradiated MOX Fuels," *PNC TN* 9410 97-054 (1997)

of the JOYO Mk-II core with a burnup of 62,500MWd/t and cooling time of 5.2 years was measured in this study. The outline of measured spent fuel subassembly is shown in Table 1.

3. Calculation

3.1 Detector Response Calculation

Because the neutron distribution was measured without collimation, neutrons emitted from the whole fuel region contributed to the counts by a B-10 proportional counter. Therefore, the detector response per neutron released from the spent fuel was required to convert the neutron counting rate into the neutron intensity. The detector response matrix as shown in Fig. 2 was calculated using the Monte Carlo code "MCNP-4A" [1)] in the water moderated condition. Some calculation conditions by "MCNP-4A" were shown in Table 2. The neutron cross section set used in this calculation was the FSXLIB[2)] processed from the JENDL-3.2 library.[3)]

As the detector counts the primary neutron from the spent fuel produced by spontaneous fission and (α, n) reactions, but also the secondary neutron which was generated by the induced fission reaction in the spent fuel with a moderated neutron. Therefore, the effect of the neutron multiplication was also considered in the calculation.

3.2 Neutron Intensity Calculation

The neutron intensity was then calculated by multiplying the atomic number of actinide nuclides by spontaneous fission and (α, n) reaction rates. The burnup calculation to obtain the change of fissile composition of U, Pu and the transmutation of minor actinides was conducted using the JOYO core management code system "MAGI".[4)] The details of the calculation method of "MAGI" is shown in Table 3. The neutron flux distribution was calculated on the basis of a 3-D diffusion theory with seven energy groups. The neutron cross section was collapsed from the 70 group JFS-3-J2 cross section set [5)] processed from the JENDL-2 library. The reactor power history and the core configuration in each operational cycle of the JOYO Mk-II core was simulated exactly in the "MAGI" calculation.

The accuracy of the neutron flux calculation was evaluated to be less than 5% in the fuel region according to the comparison between "MAGI" and reactor dosimetry test results based on the foil activation method. The axial neutron flux by "MAGI" was corrected so that the relative distribution (see Fig. 3) of calculated and measured (^{144}Pr) burnup could match.

4. Result and Discussion

Comparison of measured and calculated neutron intensity is shown in Fig. 4 and Table 4. The measured neutron intensity in the whole fuel was about 2.4×10^6 n/s, and it was about 2.5 times as much as that of fresh (unirradiated) fuel which is calculated to be about 9.7×10^5 n/s. The ratio of calculated to measured value (C/E) ranged from 0.89 to 1.47 in each fuel region with the average C/E of 1.22. The error of C/Es ranged from 14% to 32% and was mainly due to an uncertainty in solving the inverse matrix of detector response to obtain neutron intensity.

The measured ^{144}Pr distribution (Fig. 3) showed that the axial peak of burnup was slightly shifted downward due to the control rod insertion during irradiation. Different from the roughly symmetrical shape of axial burnup distribution, it was observed in Fig. 4 that the neutron intensity at the bottom of fuel region was apparently higher than that at the top. The

Neutron Intensity of Fast Reactor Spent Fuel

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Neutron intensity of spent fuel of the JOYO Mk-II core with a burnup of 62,500MWd/t and cooling time of 5.2 years was measured at the spent fuel storage pond. The measured data were compared with the calculated values based on the JOYO core management code system "MAGI", and the average C/E approximately 1.2 was obtained. It was found that the axial neutron intensity didn't simply follow the burnup distribution, and the neutron intensity was locally increased at the bottom end of the fuel region due to an accumulation of ^{244}Cm.

1. Introduction

Neutron intensity of spent fuel is important not only for the shielding design and dose evaluation of the reprocessing plant and the transportation of the mixed oxide (MOX) fuel, but also for the core management, because it contains more minor actinides than that of LWR fuel. The accuracy of neutron intensity depends on each neutron intensity from spontaneous fission and (α, n) reactions of individual nuclide, and burnup calculation for its production and transmutation. Due to the lack of accumulated data, obtaining an evaluation of neutron intensity accuracy has not been achieved yet.

In order to obtain the experimental data and to improve the accuracy of burnup calculation, the neutron intensity from a spent fuel subassembly of the experimental fast reactor JOYO Mk-II core was measured. The measured data were compared with the calculated values based on the JOYO core management code system.

2. Measurement

The neutron intensity measurement was taken in the spent fuel storage pond at JOYO as illustrated in Fig. 1. The measurement system consists of a neutron detector, a gamma-ray detector and the fuel scanning device which contains the spent fuel inside. Axial and circumferential distribution of neutron and gamma-ray emitted from a spent fuel was measured by moving the fuel scanning system vertically and by rotating itself around the fixed detectors. B-10 proportional counters with sensitivity of 0.78cps/nv_{th} or 5.83cps/nv_{th} were used for the neutron detector depending on the neutron intensity of spent fuel. Cadmium shield was covered around the neutron detector to improve the S/N ratio by reducing the thermal neutron which was moderated in the water and became the background in the measurement. Gamma-ray spectrum of the spent fuel was measured with a high purity germanium (Ge) detector to obtain burnup distribution by counting 2.186MeV peak of ^{144}Pr which is one of the representative fission products. The stainless steel collimator with lead slits was used to increase the position resolution for the gamma-ray measurement. Spent fuel