Efficiency Evaluation on A Newly Developed Neutron Counter for Small SF Sample Application

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1. Introduction

Nuclear material accounting in spent fuel handling facilities is indispensable to the safeguards and process controls [1]. Several non-destructive accounting methods are used for these purposes. In this work, counting efficiency of a neutron counter developed for a small spent fuel sample application was determined by an experimental method. And then this efficiency was compared and evaluated with that of the MCNPX code calculation.

2. Experimental

2. 1. Neutron Counter Setup

A newly developed neutron counter is consisted of two major parts, that is, main body and electronic equipment. Main body has six He-3 detectors, high density polyethylene (HDPE) neutron moderator, gamma ray shielding material, sample hole, PDT unit, cover/LED panel and so on. A spent fuel sample holder and sample holder supporter are used to be positioned at the center of neutron counter. And electronic equipment is consisted of a shift resister (JSR-14) and a personal computer loaded with INCC software. Fig.1 shows a schematic drawing of neutron counter for small spent fuel sample application.

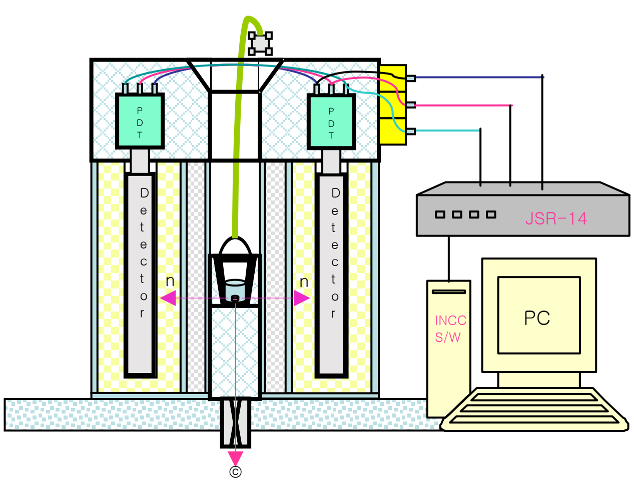

Fig.1. Schematic Drawing of Neutron Counter.

2.2. Efficiency Determination

Efficiency of neutron counter is determined by the equation (1).

$$\varepsilon = \frac{measured\ neutron\ count\ rate\ (c/s)}{s\tan dard\ neutron\ source\ \mathrm{int}ensity\ (n/s)} \tag{1}$$

Cf-252 neutron source was used as a standard source, and its intensity at present was calculated by considering the elapsed time and half life, initial intensity, etc as listed in Table 1. As a result of calculation, source intensity is 816 n/s (20,240 n/s in 1999). And neutron count rate obtained from the experiment with the neutron counter and standard source is 92 c/s. So the efficiency of newly developed neutron counter was determined by inserting these values in the equation (1), 11.1%. Therefore, it was confirmed that this value is almost the same as that of the MCNPX code calculation carried out for this neutron counter design.

Table 1. Source information and efficiency results determined by the experiment and MCNPX code calculation

Cf-252 intensity		Half life	Elapsed time	Neutron count rate		Efficiency	
1999.2.	2008.9			total	backgrd	MCNPX	Experiment
20,240n/s	816 n/s	2.646 yr	8.5 yr	92.0 c/s	0.39 c/s	10.7 %	11.1 %

2.3. Future Work

A shielded glove-box as shown in Fig.2 for the installation of gamma detection system and neutron counter has been partially modified. And a cask adapter connected on the sample inlet/outlet door was fabricated to transfer the spent fuel samples. A newly developed neutron counter will be installed in the glove-box in the near future for the small spent fuel sample application.

3. Conclusions

Efficiency of a newly developed neutron counter was determined by an experimental method, it was confirmed that this value is almost the same as that of the MCNPX code calculation carried out for this neutron counter design.

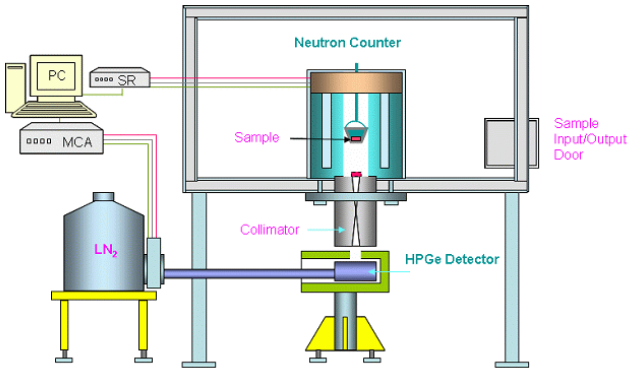

Fig.2. Schematic drawing of shielded glove-box for gamma and neutron counter

REFERENCES

[1] Wayne D. Ruhter, R. Stephen Lee, Herbert Ottmar and Sergio Guardini, "Nondestructive assay measurements applied to reprocessing plants", Proceedings of the tripartite seminar on Nuclear Material Accounting and Control at Radiochemical Plants, Obninsk, Russia, 135-155(1988).