Hot Test Results of the ACP Safeguards Neutron Counter with Spent Fuel

Tae-Hoon Lee*, Seung-Hyun Kim, June-Sik Ju, Hee-Sung Shin, Ho-Dong Kim
Korea Atomic Energy Research Institute, Yuseong, Daejeon 303-600, Korea
**Corresponding author: typhoon@kaeri.re.kr*

1. Introduction

A passive neutron coincidence counter for the nuclear material measurement of the Advanced Spent Fuel Conditioning Process (ACP) has been developed by the Korea Atomic Energy Research Institute (KAERI) since 2003 and was deployed in a hot cell of the ACP Facility (ACPF) in 2005 [1]. The most dominant neutron source among the spontaneous fission nuclides contained in spent fuel is ^{244}Cm. To obtain the neutron counting rates of the singles, doubles, and triples coincidences of the neutron counter with an increment of the ^{244}Cm mass, a hot test of the neutron counter was performed with several spent fuel rod-cuts at the ACPF hot cell in 2007. The destructive assay (DA) results of the spent fuel samples were obtained in 2008. This study shows the high performance and reliability of the ASNC for ^{244}Cm measurements of spent fuel. The measurement and MCNPX code simulation results for ^{244}Cm neutron counting are presented and also a comparison with DA results will be shown.

2. Methods and Results

The ACP Safeguards Neutron Counter (ASNC) has been developed as an instrument to quantify the fissile elements contained in the ACP process materials. This neutron counter was designed to have a full-remote maintenance capability for an operation in a hot cell environment. The neutron detection efficiency of the ASNC is about 21% which was measured with ^{252}Cf neutron sources. Other various important characteristic parameters of the ASNC except its operating high voltage (HV) were also measured with the ^{252}Cf sources.

2.1 Signals from Process Materials

The target materials for the ACPF safeguards are spent fuel rod-cuts (UO_2), U_3O_8 powder, electrolytic-reduced uranium metal and hulls. All these materials are containing fission products emitting intensive gamma-rays and thus the gamma pileup signals interfere with spontaneous fission neutron signals whose neutrons are emitted by even-number actinide isotopes. Thus the counting signals caused by the gamma pileup should be prevented by adopting a proper operating HV of the neutron counter. There are various spontaneous fission neutron sources such as ^{238}U, ^{238}Pu, ^{240}Pu, ^{242}Pu, ^{242}Cm and ^{244}Cm in the ACP safeguards target materials and the most dominant neutron source among those isotopes is ^{244}Cm [2]

2.2 A Hot Test of the ASNC

A hot test of the ASNC was performed at the ACPF hot cell with several spent fuel rod-cuts in 2007. The purpose of the test was to check the gamma pileup and to set the operating HV to a proper value. Another reason for the hot test was to calibrate the ASNC for ^{244}Cm [3]. The hot test had been performed for almost one month in cooperation with two experts from the Los Alamos National Lab. (LANL) and the International Atomic Energy Agency (IAEA). The HV plateau curves of the ASNC for singles and doubles rates were obtained as changing the number of rod-cuts so with an increment of the number of gamma-rays as well as neutrons. The gamma pileup was clearly suppressed by setting the operating HV to 1680 V just below the plateau knee as shown in Fig. 1. The neutron counting rates of the ASNC were also measured as increasing the number of spent fuel rod-cuts so the calibration curves for ^{244}Cm were obtained for singles and doubles rates, respectively.

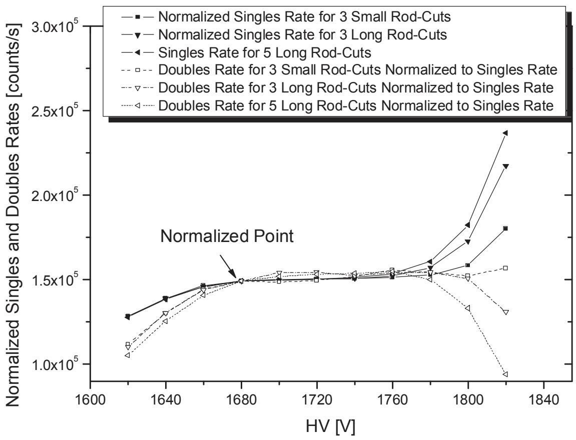

Fig. 1. HV plateau of the ASNC for the spent fuel rod-cut.

The MCNPX code simulation was also performed to calculate the neutron counting rates of the ASNC for ^{244}Cm of spent fuel rod-cuts. The source term of the spent fuel was obtained by using the ORIGEN-ARP burnup simulation code. However, there was a difference of around 10% in neutron counting rates between measurements and MCNPX simulation results. On the other hand the ratios of singles to doubles rates and doubles to triples rates for the measurements were almost the same with those for the MCNPX results. Since the burnup information provided by the nuclear power plant (NPP) could be incorrect, the source term results obtained by the ORGEN-ARP code simulation with the declared burnup data from the NPP could be different from true values.

2.3 Chemical Analysis Results of Spent Fuel

To obtain the true burnup of spent fuel rod-cuts, their small samples were sent to the chemical analysis laboratory at KAERI in 2007. The Pu and U contents can also be obtained as the results of the destructive assay (DA). The DA results of the samples were acquired last year. The difference in the burnup information between the declared and the DA results was around 10% as shown in Fig. 2 and the reason for the discrepancy between measurements and MCNPX results can be explained from this difference. The expected burnup of each rod-cut was also obtained from the results of self-multiplication correction for the measured ^{244}Cm mass of the rod-cuts and the results coincide with the DA results in less than 2% difference as shown in Table I [4]. The calibration curve of the ASNC for ^{244}Cm measurements was obtained and the calibration constant is 297.0 [doubles counts/s/mg ^{244}Cm] as shown in Fig. 3.

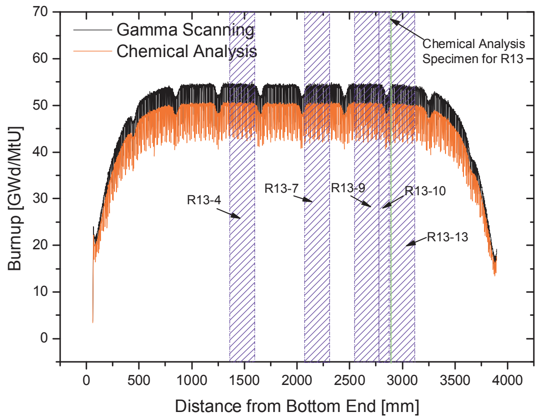

Fig. 2. Gamma scanning profile of a spent fuel rod.

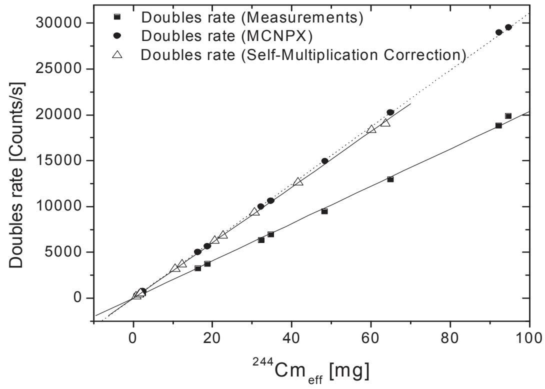

Fig. 3. ASNC's calibration curve for ^{244}Cm of spent fuel.

3. Conclusions

This study shows a high performance of the ASNC for the ^{244}Cm measurements of spent fuel and also the burnup of spent fuel samples can be obtained through a series of ORIGEN-ARP code simulations by comparing the ASNC's measurement results for ^{244}Cm mass with the source term of the ORIGEN-ARP code simulation results.

REFERENCES

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Table I: Cm measurement results of the ASNC and DA results for the spent fuel rod-cuts

Rod	ID	Length (cm)	Burnup (GWd/MtU)			(ASNC-DA)/DA			
			NPP-Declared	Cm Measurement Based (ASNC)	Chemical Assay (DA)	Burnup	Cm	Pu	U-235
R13	4	25	53.465	49.072	49.453	-0.77%	-3.48%	-0.33%	1.49%
	7	25	53.277	48.814	49.278	-0.94%	-4.23%	-0.40%	1.82%
	9	25	52.984	48.440	49.007	-1.16%	-5.16%	-0.50%	2.23%
	10	1	51.330	47.963	47.478	1.02%	4.85%	0.45%	-1.85%
	13	25	53.055	48.389	49.073	-1.39%	-6.22%	-0.60%	2.69%
C16	7	25	59.905	55.592	56.081	-0.87%	-3.82%	-0.30%	1.99%
	10	1	58.212	54.823	54.494	0.60%	2.76%	0.24%	-1.33%
	12	1	58.787	54.017	55.032	-1.84%	-7.91%	-0.72%	4.17%