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Development of Neutron Detection System for CANDU Spent Fuel Verification

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1. Introduction

Improvement of spent fuel verification technique and instrumentation is one of the significant challenge in the safeguards field. The CANDU spent fuels stored in the storage pond must be transported to dry storage for the long term storage. A dry storage canister is loaded with 9 baskets and each basket contains 60 spent fuel bundles. This corresponds to 34.9 kg of plutonium (1 SQ = 8kg for Pu) per each canister[1]. After the canister is fully loaded, “gamma-ray fingerprinting” is currently being implemented to compare a pre-established baseline measurement against subsequent measurements in particular case of a Containment/Surveillance failure. However, IAEA still lacks effective re-verification method for spent fuel dry storage.

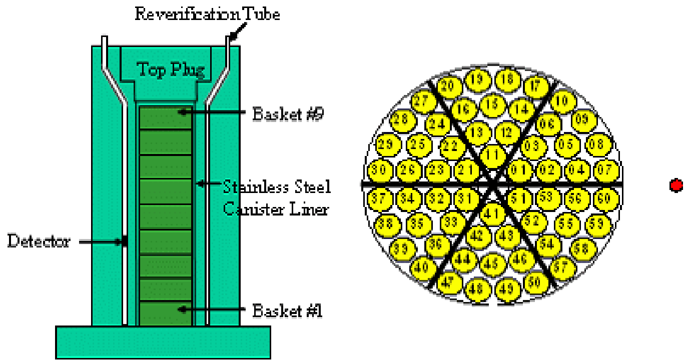
Neutron detection has been proposed to use as verification and re-verification methods for spent fuel dry storage. A neutron detection system can provide for better visibility of the canister contents than a gamma ray detection system because the neutron has good penetration in the high density, high Z materials of reactor fuel comparing to the gamma ray characteristic. In this study, detailed MCNP model of the canister was constructed and the neutron detection system was designed for the feasibility test.

2. Simulation

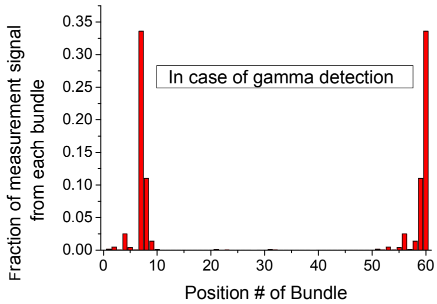
The detailed MCNP model was construct consisting of concrete structure, 9 fuel basket and 540 fuel bundles. Two re-verification tube are place on opposite sides of the cask at distance of 73.66 cm from the central axis of the canister. Figure 1 (a) shows side view of CANDU spent fuel canister and the bundles array in spent fuel basket.

CANDU spent fuel source term was calculated using ORIGEN-ARP[2]. The reference spent fuel was typical CANDU spent fuel of 7500 MWd/MTU burn up and 10 years cooling time. The total neutron production rate per metric ton is calculated to be 3.13×10^{6} n/s and total gamma ray intensity is estimated to be 1.2×10^{16} .

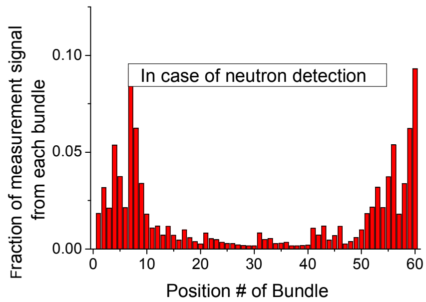
The contributed fraction of the gamma and neutron signal from each bundles in a basket were calculated using MCNPX code[3], and the results are shown in figure 1 (b) and (c) respectively. The detector is positioned at the center of a 4^{th} basket in the re-verification tube. The source terms are activated for the 4^{th} basket only, therefore the result does not reflect the effect from other baskets. In regard to neutrons, an expected count rate (count/s) for He-3 tube(7.5 atm, 1 inch diameter and 2 inch active length) from each bundle was calculated, whereas in regard to gamma rays, an expected dose rate(R/h) on a detector was calculated. In the calculation, the gamma detector was similar in size and shape to the He-3 tube and located at the same position in re-verification tube as the neutron detector.



(a)



(b)



(c)

Figure 1.(a) Side view of CANDU spent fuel canister and top view of the basket, (b) Fraction of gamma signal

induced by an individual bundle, (c) Fraction of neutron signal induced by an individual bundle

3. Neutron detection system

3.1 System configuration

The neutron detection system was designed to pass trough a bend on the re-verification tube. A detector connected to the preamplifier as shown in figure 2. To traverse a bend, the solid part would be smaller than 7.1 inch length and 1.3 inch diameter. He-3(7.5 atm with 1inch diameter and 2inch active length) and BF3 (600 torr with 1 inch diameter and 2 inch active length) detectors were used for the test.

In strong gamma fields, the detectors may decrease neutron counting ability due to the effect of gamma pulse pileup. Therefore the operating high voltage should be determined with an actual spent fuel canister to verify the gamma ray pileup and to obtain maximum neutron count rate.

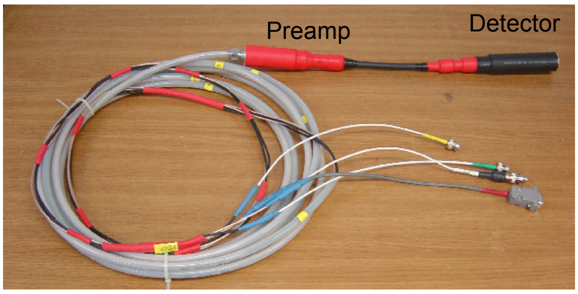
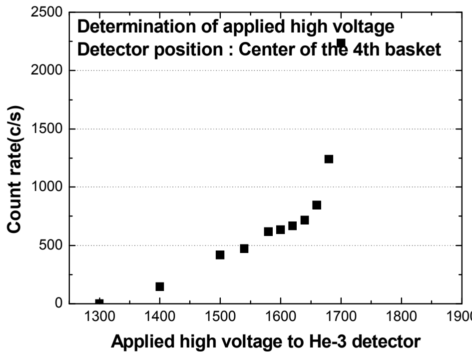


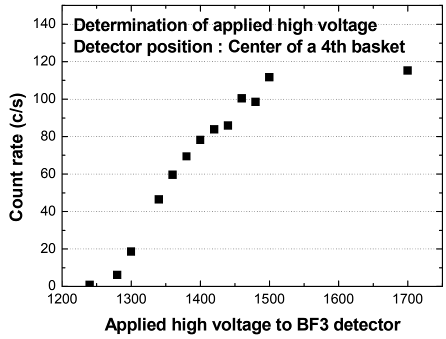
Figure 2. Photo of a neutron detection system.

3.2 Determination of high voltage

In order to verify gamma ray pileup and to determine the applied high voltage, the neutron counting systems were tested at the actual spent fuel canister. The detector was positioned at the center of 4th baskets in the re-verification tube and measured count rate as applied voltage. As shown in figure 3, typical gamma pileup characteristic was found for He-3 detector, while there was no visible gamma effect on BF3 tube. This might be due to the difference of gas pressure and Q-value of the detectors.



(a)



(b)

Figure 3. Detector count rate with respect to the applied high voltage (a) He-3 detector having 1 inch dia. and 2 inch active length with 7.5 atm gas pressure (b) BF3 detector having 1 inch dia. and 2 inch active length with 600 torr gas pressure.

4. Summary

Neutron detection was proposed to use verify CANDU spent fuel canister in case of a Containment/Surveillance failure. The signal contribution from individual bundles to the detector was calculated for the neutron and gamma detection. From the calculation, it is expected that the passive neutron detection provide entire spent fuel information in the canister in reasonable measurement time.

The prototype neutron detection systems were fabricated using He-3 and BF3 detectors, and the preliminary field test was performed in this study. Total neutron count rate was measured and the operating voltage was determined in order to achieve maximum count rate in the real field. For He-3 detector, the detector operating voltage was determined around 1640V and the count rate was 736 c/s. It seems that the gamma signal is dominant over 1750 V due to the gamma pileup effect. For the BF3 detector, the gamma pileup effect was not visible until applying the H.V. up to 1700V. The neutron count rate of the BF3 detector was measured 111 c/s at 1500V. In the future, the reproducibility and stability tests will be performed, and the detection limit and actual validity of the neutron detection system for the re-verification of the canister will be discussed.

REFERENCES

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[3] HENDRICKS, J.S. ed., MCNPX, Version 2.6.A, LA-UR-05-8225, 2005, Los Alamos National Laboratory.