Neutron Monitor Development for Spent Fuel Surveillance in Hotcell

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Abstract

Neutron monitor has been developed and simulated for the possible use in determination of spent fuel quantity and movement in hotcell. This is necessary and practical for the purpose of safeguards implementation and control of process line under high radiation hotcell area. The count rates for neutron monitor due to spontaneous fission neutrons from spent fuel is calculated by using the MCNP model. The effects of neutron counts on burnup and location of fuel rod are investigated. It shows that the calculated count rates of neutron monitor agree well that of measured count rates with PWR spent fuel rod in hotcell.

1. Introduction

The neutron measurement method is an important role that checks the special nuclear material in nuclear facility. The new neutron monitor system has been developed to survey the quantitative needs of spent fuel in a hotcell under high radiation level.

Spent fuel contains the heavy metal nuclides such as uranium, plutonium and curium called as actinides plus their daughters. The primary emission neutrons originate from the spontaneous fission of curium and plutonium as well as some (α ,

n) reaction neutrons from the fuel materials. It is difficult to examine the contents, location and characteristics of spent fuel materials. The neutron monitoring system used as neutron counting has been investigated to measure the location and contents of spent fuel material under safeguards technology development [1-4]. However, the existing gamma-ray measurement method could be not appropriate to determine the unknown geometry and characteristics of spent fuel in hotcell due to high gamma radiation level.

In order to survey the movement and characteristics of spent fuel in hotcell, the He-3 neutron counting method has been studied by applying a neutron monitor. The neutron monitor is used in total neutron counting mode because the signal to be measured is often weak. Total neutron counting method are usually unsophisticated instruments that do not measure neutron energy directly. The Monte Carlo method can allow a detailed three-dimensional geometrical model to be constructed mathematically to simulate a physical model. The MCNP code[5] has been applied to design the neutron monitor model.

2. Neutron counting method

The neutron measurement instrument for total neutron counting use moderated He-3 detectors. The He-3 detectors are relatively simple to operate and their reliability is good. They can tolerate approximately 10^{13} fast n/cm^2 without serious radiation damage and they can provide adequate discrimination against gamma rays in field less than 1 R/h.

The Bohnel point model equations[2] provide a means of predicting an observed neutron count rate from spent fuel. The point equation for total count rate(singles count rate) is given by

$$\dot{S} = \varepsilon_t \, M_L \, \sum_n (SF)_n (1 + \alpha) \qquad (1)$$

where,
  $\dot{S} = $ singles count rate (total rate)
  $\varepsilon_t = $ the total detector efficiency
  $M_L = $ leakage multiplication of sample
  $SF_n = (mF)_n (\nu_{s1})_n$, spontaneous fission rate of isotope n.(fissions/sec)
  $m = $ sample mass
  $F = $ fission rate
  $(\nu_{s1})_n = $ 1st reduced spontaneous fission moment of isotope n.(neutrons/sp. fission)
  $\alpha = $ ratio of (alpha, n) emission to spontaneous fission

The spontaneous fission and (α,n) neutron source terms are dependent on fuel burnup and initial enrichment. The dominant source term of neutrons is spontaneous fission from ^{244}Cm in spent fuel. Self-multiplication within a fuel sample increase the total neutron count defined the net leakage multiplication which is caused in terms of induced fission in fissile material.

3. Test model for neutron monitor

For a particular neutron detector design, detection efficiency depends on the incident neutron energy because of the dependence of the ^{3}He(n,p) cross section on neutron energy and the moderating effect of hydrogen in polyethylene. In general, ^{3}He(n,p) cross section at thermal neutron is higher than that of fast neutron. The neutron monitor around spent nuclear fuel in hotcell was not easy to measure because of the high gamma-ray backgrounds. In order to obtain reasonable detector efficiency, a proper shield layer is needed to protect the He-3 tubes and preamp(PDT) from high gamma dose.

The model was to develop the MCNP code simulation of capability to measure the neutron counts from neutron emissions of spent fuel. With a fixed number and arrangement of He-3 tube counters, the amount and location of polyethylene moderator can strongly influence counting efficiency. Figure 1 shows a horizontal and vertical cross section of the MCNP simulation model for neutron monitor. The neutron counter in the cavity are composed of two He-3 tubes with 50 cm in active length and 2.5 cm in diameter. The neutron counter tubes have approximately 50 cm long enough to get the higher efficiency for spent fuel rod. The polyethylene layer as moderator is placed around two He-3 tubes. The poly encased with stainless steel shell has 2 holes for He-3 detector tubes which can detect neutrons by (n, p) reaction.

The thick lead layer gives gamma-ray shielding protecting the He-3 tubes and preamp from high gamma radiation level. The radiation source term of the spent fuel is defined for the multigroup shielding analysis. For the typical PWR in equilibrium, fresh fuel at 3.2% U-235 enrichment is considered. The ORIGEN2 code used to calculate the buildup of the fission products, activation products and actinides during the irradiation. The shielding analyses were performed by MCNP-4B code. The lead shield thickness is 4 cm from polyethylene layer to outer stainless steel plate. The thickness of lead is determined by shielding analysis to reduce radiation level to approximately 1,000 mR/h at neutron counter. The neutron monitor located in hotcell is

shown in figure 2. The spent fuel rod will move from front side to other direction at lower position of neutron monitor. The test fuel rod sample selects a PWR fuel rod having a burnup of 39 GWd/tU and a cooling time of 6.1 years.

4. Results and Discussions

    The neutron monitoring system has been investigated to measure the location and contents of spent fuel material under safeguards technology development. The existing neutron counter could be appropriate to determine the unknown geometry and characteristics of spent fuel in hotcell under high gamma radiation level. In order to survey the movement and characteristics of spent fuel in hotcell, the neutron counting method has been studied by applying a neutron monitor.

    Figure 3 shows the total count rates dependent on fuel burnup to 40 Gwd/Mtu. The count rates increase that the rod length insert in hotcell. And, also the neutron count rates increase as higher burnup. This explains that the spent fuel with high burnup contain higher neutron sources. The calculated count rates are compared with experimental values. To measure the neutron counts at neutron monitor in hotcell, the neutron monitor is shown in Figure 4. The total neutron count rates for comparison with measured data is shown in Figure 5. The measured neutron count rates increase with rod insertion length. The measured values agree well with that of the calculation. The present study shows that the neutron monitor can be applied for surveying the spent fuel in hotcell.

5. Conclusion

    The neutron monitoring system in hotcell has been investigated to measure the location and contents of spent fuel material under safeguard technology development. The neutron monitor can be applied to the prediction of spent fuel movement and characteristics in hotcell. To enhance accuracy of the measurement method for predicting the spent fuel behavior in nuclear facility, the neutron monitor will be continually developed by further study.

References

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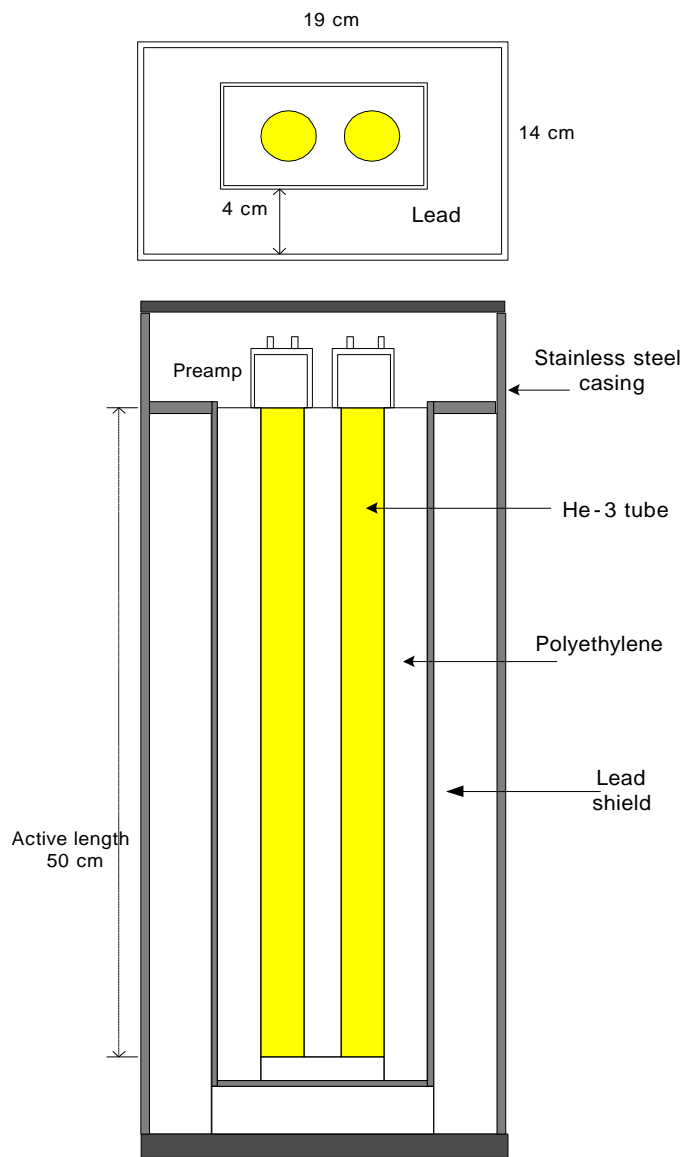
19 cm

14 cm

4 cm

Lead

Preamp

Stainless steel
casing

He- 3 tube

Polyethylene

Lead
shield

Active length
50 cm

Figure 1 Neutron monitor model for spent fuel in hotcell

He- 3 tube

Polyethylene
Moderator

Lead shield

Spent
fuel rod

Neutron
Monitor
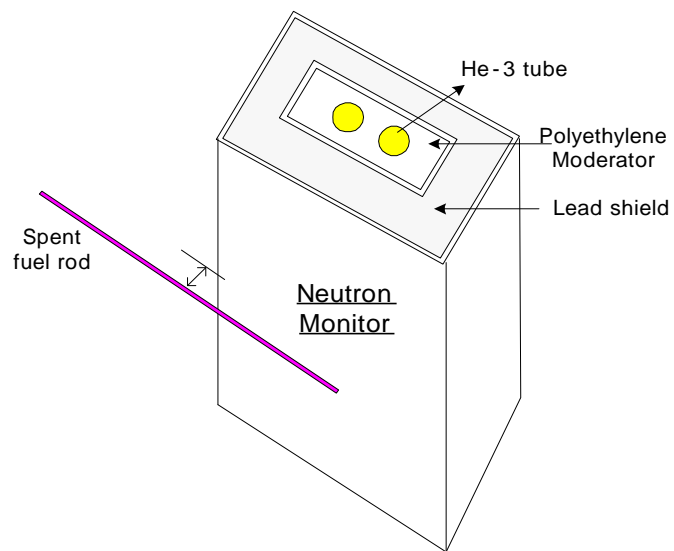
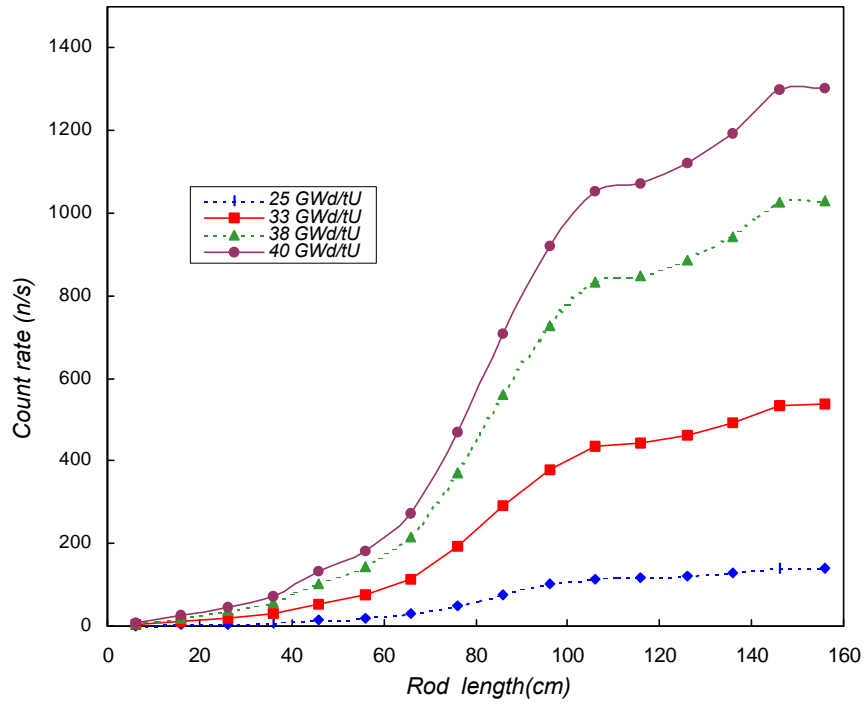Figure 2   Neutron monitor location with
spent fuel rod in hot cell

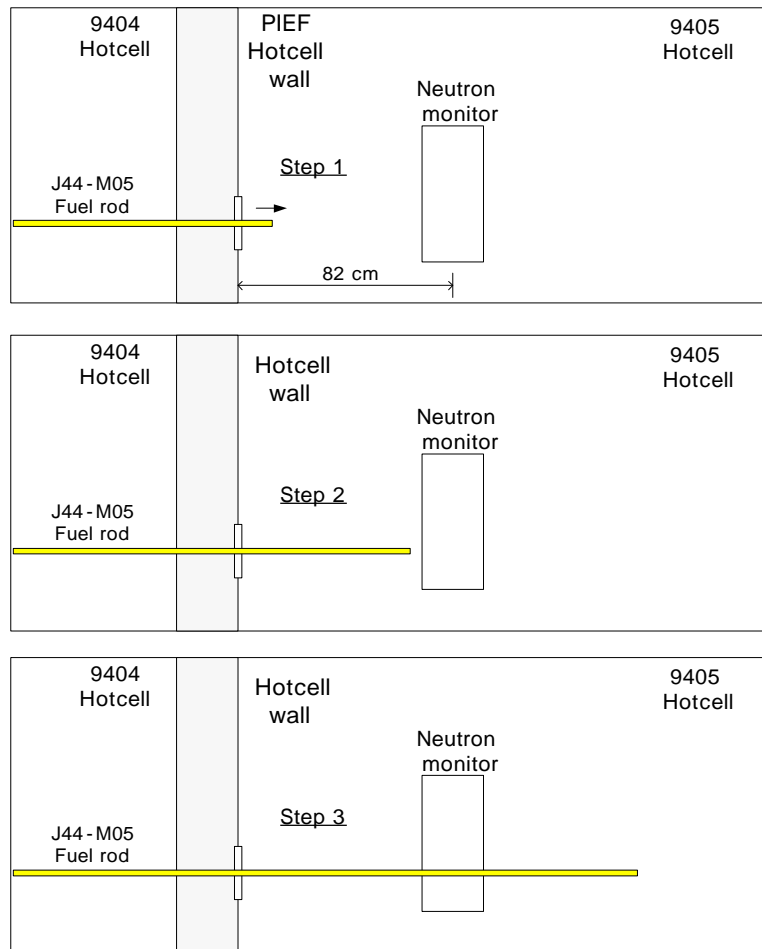Figure 3   Total count rates dependent on fuel burnup versus
rod length in hotcell

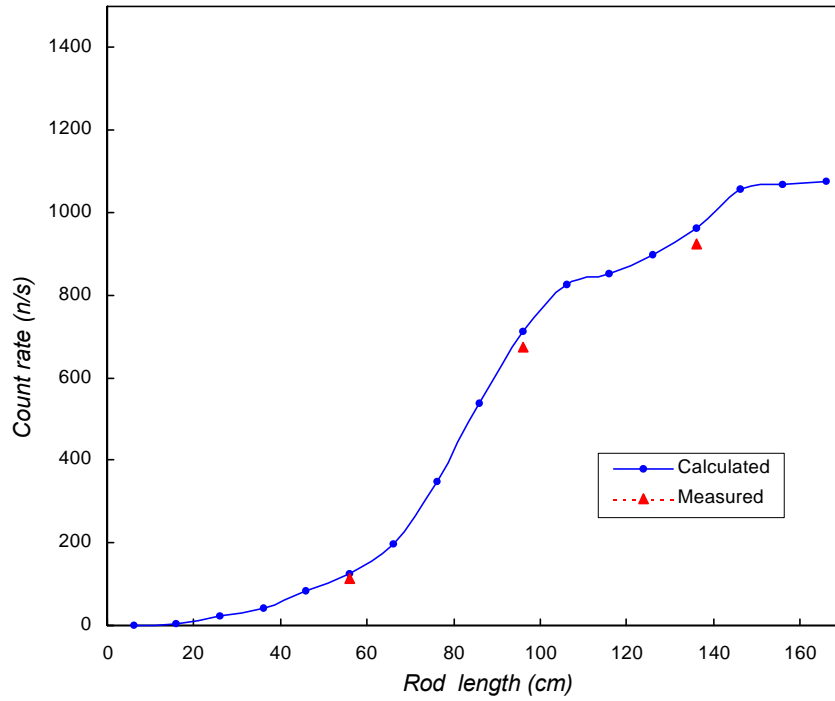**Figure 4** Experimental arrangement of neutron monitor in PIEF hotcell

Figure 5   Comparison of count rates between the measured
and calculated values versus rod length in hotcell