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Feasibility Study of New Spent Fuel Verification System at LWR

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1. Introduction

In case of light water reactors (LWR), verification of the contents of spent nuclear fuel assemblies is one of the basic safeguards measures routinely carried out by authorities and inspectorates. The basic objective is to gain assurance that the operator declared data concerning isotopic contents and mass are correct. In addition to correctness, also completeness of the data needs to be verified to gain assurance that no material is missing.

In Korea, one of the verification measurements was carried out to detect the gross defect using SFAT (Spent Fuel Attribute Tester) to measure gamma signatures from fuel assemblies stored in the wet storage of LWR. Gross defect level measurements result in a conclusion whether the assembly verified is completely missing or replaced with a dummy. However, the increased long term spent fuel storing and the changed fuel design made more difficult to verify gross detect. So, new verification system is required using neutron which is more penetrable than gamma.

The present study aims at feasibility study of new LWR spent fuel verification system using the MCNPX that is a well-known and widely-used Monte Carlo radiation transport code [1], which may be useful in the verification measures of the LWR spent fuel assembly.

2. Methods and Results

2.1 Source Term

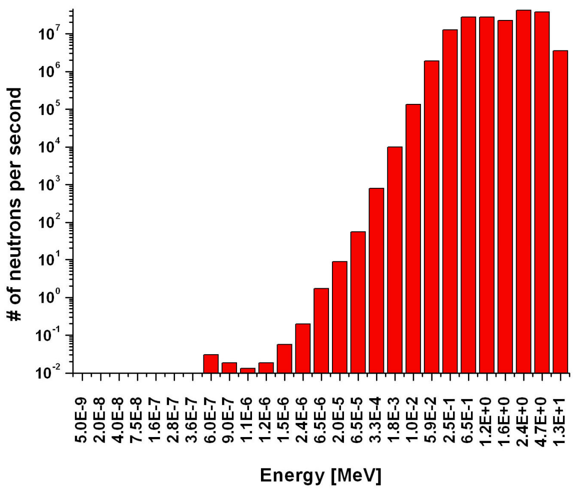

Fig. 1. Neutron source spectrum per assembly for PWR spent fuel of 40900 MWd/MTU burnup and 15 years of cooling time calculated by ORIGEN-ARP.

The material composition and total neutron production spectrum of the LWR spent fuel assembly were evaluated using the fuel depletion code ORIGEN-ARP[2] with following assumptions: (a) All assemblies are the same as PWR reactor's spent fuel of 40900 MWd/MTU burn-up and 15 years of cooling time. The neutron production spectrum (# of neutron/s) per assembly is shown in Fig. 1. The integrated neutron intensity is 1.759×10^8 #/s per assembly.

2.2 Simulation

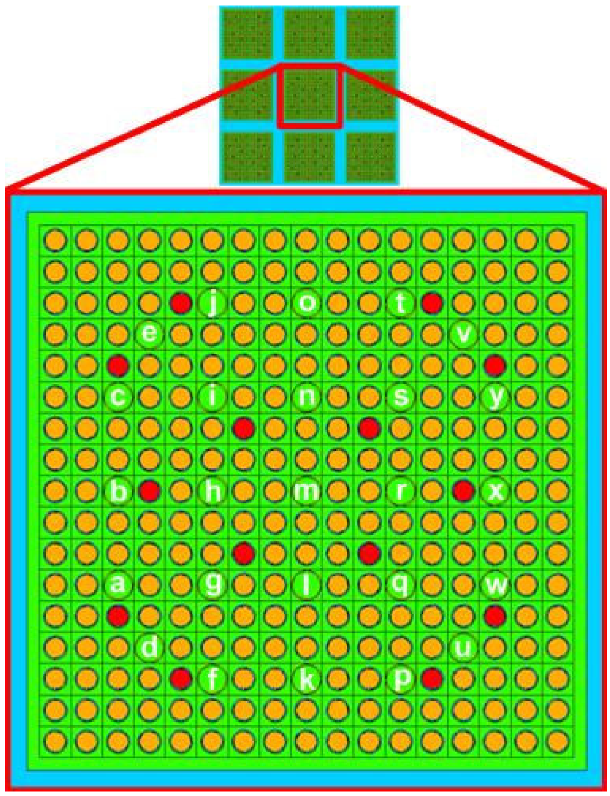

Fig. 2. Cross sectional view of the total simulation geometry.

Fluxes at the guiding tube in the spent fuel assembly were calculated using the Monte Carlo code system MCNPX^{TM}, version 2.5 with the flux tally F4. Four possible cases, real spent fuel, low burn-up spent fuel, dummy, and no material in the fuel rod, were simulated to evaluate the potentialities. The dimensions and materials of the assembly provided by the manufacturer were inserted with following assumptions: (a) The dimensions of the individual fuel rods were not changed during irradiation, (b) The integrated neutron intensity is 1.0×10^6 #/s per assembly for time consuming, (c) The dummy fuels were made of stainless steel. The detailed model of assembly was constructed consisting of 264 fuel rods and 25 guide tubes. And the total assemblies were modeled nine in the water. The radius of the fuel pellets is 0.4096 cm. The radius of the inner and the outer radius of the cladding are 0.4178 cm and 0.475 cm. The fuel pin pitch is 1.2598 cm. The overall fuel rod length is 409.3667 cm and the pitch of the storage rack is 27.4 cm. The cross sectional view of the total simulation geometry is shown in Fig. 2.

Figure 3 shows the neutron flux in the guide tube corresponding to the each position in each case. It can be seen that the fluxes of the neutron were decreased to ~45% of the neutron flux with real spent fuel for low burn-up and empty. In case of dummy, the neutron flux was decreased to ~70%. Based on this result, it was found that new verification system using neutron measurement has adequate performance to find the gross detect.

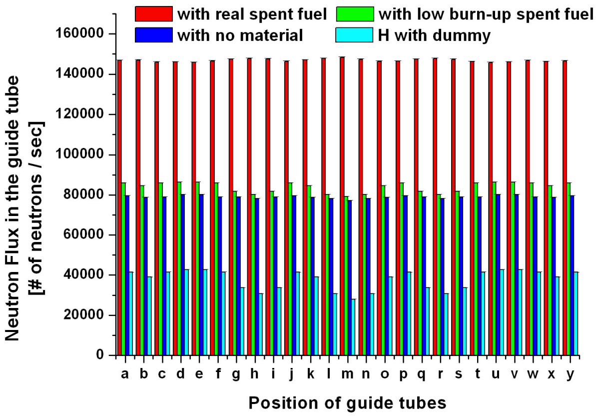

Fig. 3. Calculated neutron flux in the guide tube with real, low burn-up, empty, and dummy spent fuel.

3. Conclusion

In this study, ORIGEN-ARP and MCNPX calculations were performed to establish the expected neutron source term from the PWR spent fuel and to study the new verification system for detection of gross defect, respectively. It can be seen that the fluxes of the neutron were decreased to ~45% of the neutron flux with real spent fuel for low burn-up and empty. In case of dummy, the neutron flux was decreased to ~70%. Based on this result, it was found that new verification system using neutron measurement has adequate performance to detect the gross detect.

REFERENCES

[1] LANL, *MCNPX^{TM} User's Manual*, Los Alamos National Laboratory, MCNP^{TM} 2.5.0 LA-CP-05-0369, 2005.

[2] I.C.Gauld, ed.,″ORIGEN-ARP : Automatic Rapid Processing for Spent Fuel Depletion, Decay, and Source Term Analysis, ″ Oak Ridge National Laboratory, ORNL/TM-2005/39, 2005.