Computational Safeguards Analysis of PWR Spent Nuclear Fuel in Dry Storage – 17435

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ABSTRACT

The master plan for spent nuclear fuel (SNF) management in Korea recently adopted by Korean atomic energy promotion council. According to the master plan, the detailed practical and R&D strategies for SNF transferring, storage and disposal should be determined in the near future. Besides, it is also necessary to make plan how to secure SNF at each step in terms of nuclear safeguards, safety, security and nonproliferation.

Several nondestructive analysis (NDA) techniques have been studied for assuring the safety of SNF in PWR methods. Obviously, gamma-ray is most dominant radiation emitted by SNF so several gamma-ray measurement methods have been developed and applied for the purpose of SNF monitoring. Those methods, however, suffer from a lack of nuclear data and interfering background radiation.

To date, several neutron measurement methods have been developed as a solution to the problem of gamma-ray methods. Korea Institute of Nuclear Nonproliferation and Control (KINAC) currently develops a new novel technique of fast neutron detection for spent fuel monitoring at dry spent fuel storage facilities. As the first step, a computational program has been developed in order to evaluate neutron flux on the surface of PWR dry storage cask and study for some potential sabotage events.

Typically, neutron flux on the surface of dry storage cask can be evaluated via two steps. The source term, based on burn-up rate of SNF, its cooling time and others related to operation condition at nuclear power plants, can be firstly defined by ORIGEN code. The surface flux can be then evaluated via MCNP code. These two separated steps are complicated and time consuming so KINAC developed a new algorithm and user-interface, coupled ORIGEN and MCNP.

INTRODUCTION

With a purpose of verifying that special nuclear containment in cask or storage is maintained, KINAC is currently developing a new method of passively scanning a spent fuel cask or storage using pressurized He-4 gas scintillation fast neutron detectors for nuclear security and safeguards applications. Useful unambiguous information of spent fuel integrity or discharge (sabotage events) will be provided through this project.

Neutron flux on the surface of a PWR dry storage cask were first calculated using ORIGEN and MCNP codes. Neutron release rate and energy spectra from a SNF were calculated by using ORIGEN-ARP code. The neutron flux on the surface of a PWR dry storage cask was then evaluated by using MCNP code based on geometric and material properties of the spent fuel assemblies and casks.

Both Origen and MCNP codes are so powerful tools and widely used for calculating the buildup, decay, and processing of radioactive materials and neutron, photon, electron, or coupled particle transport [1, 2]. However, they do not have a graphical user interface (GUI) so it is not easy to write input and understand output for codes. Besides, ORIGEN output files should be reformatted for neutron transport calculation via MCNP in order to evaluate neutron flux from spent nuclear fuel. This separated two step are too much complicate and time consuming so KINAC developed a new algorithm and GUI, coupled ORIGEN-ARP and MCNP, and has been studied for some potential sabotage events.

A NEW SPENT FUEL CASK MONITORING SYSTEM

This project will (1) create a neutron spectrometer using He-4 gas scintillation fast neutron detectors through an experimentally validated neutron energy response function and spectral unfolding techniques, (2) leverage test measurements and a computational spent nuclear fuel library enabling the analytical design of a cask (storage) measuring prototype system, (3) conduct a prototype instrument and employ it to monitor a nuclear spent fuel cask or storage at a commercial nuclear power reactor facilities, (4) use spectral-specific measurement data to show proof of concept for measurable data successfully converted to quantifiable signatures, such as diversion of fuel, and maintenance of fuel and cask integrity of fuel, and (5) develop a cask imaging capability. This research will mainly be carried out through the following logical path to work accomplishment.

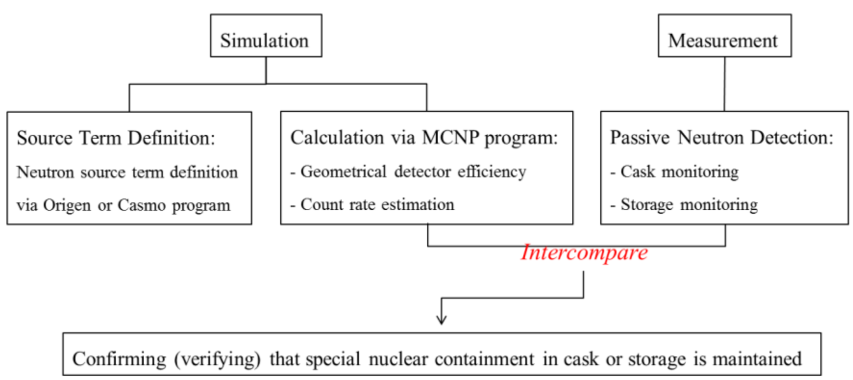

Fig. 1. The logical path to work accomplishment in this study

A new experimental apparatus, employing some He-4 scintillation detectors, will be designed and applied.

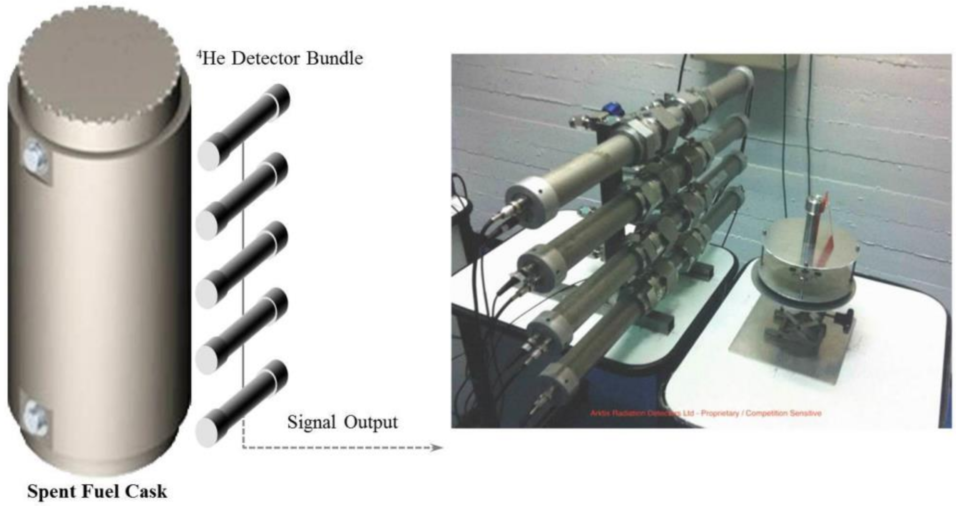

Fig. 2. The concept of a new experimental apparatus

He-4 scintillation detectors are vertically daisy-chained and the output signal can be analyzed individually from each detector or collectively from these daisy-chained detectors working as a single detector.

COMPUTATION ANALYSIS

As shown in Fig. 1, neutron flux on the surface of a PWR dry storage cask should be first evaluated in order to compare with experimental results. Thus, we developed a new algorithm and user-interface, coupled ORIGEN and MCNP codes.

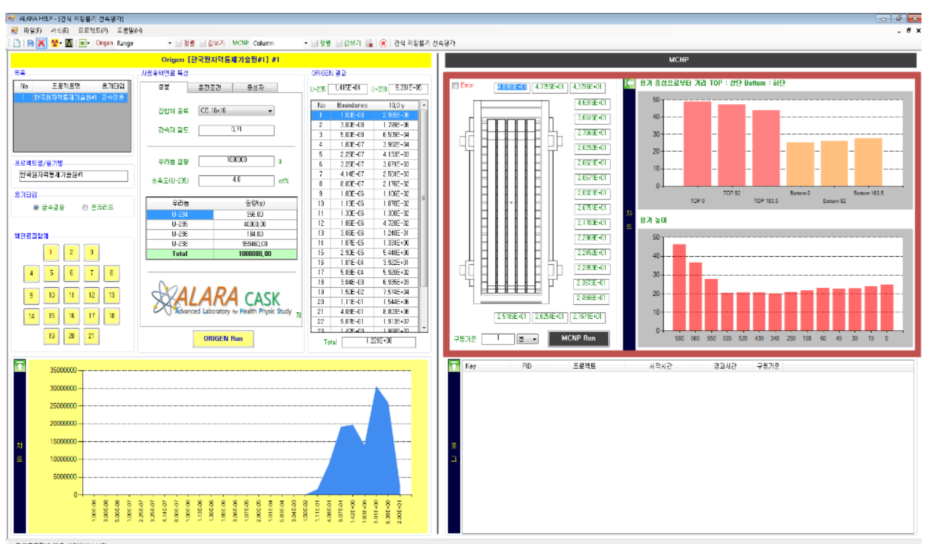

Fig. 3. A user-interface KINAC currently developed

The input parameters of the program are assembly type (CE 16x16, WH 14x14, WH 17x17), initial enrichment rate, burn-up rate, cooling time and cask type (concrete or metal). All parameters were decided based on the radiological characteristic of Korea nuclear fuels and the structural properties of a cask, newly developed by Korea Radioactive Waste Agency (KORAD) and called KN-21 [3]. KN-21 is designed to contain 21 spent nuclear fuel assemblies and be able to use for interim storage and transportation.

Neutron release rate and energy spectra from a SNF is calculated via ORIGEN-ARP and the results obtained by ORIGEN-ARP are automatically reformatted for MCNP input files by a new algorithm. This program is featured by its enhanced usability. A user can easily evaluate neutron flux on the surface of the cask by one-click application. This program also allows that a user enters parameters for each assembly. It means that some sabotage events can be analyzed through this program.

In order to know the changes of the neutron flux with some sabotage events, we assumed that some assemblies are stolen like Fig. 4. The empty spot was indicated by grey color.

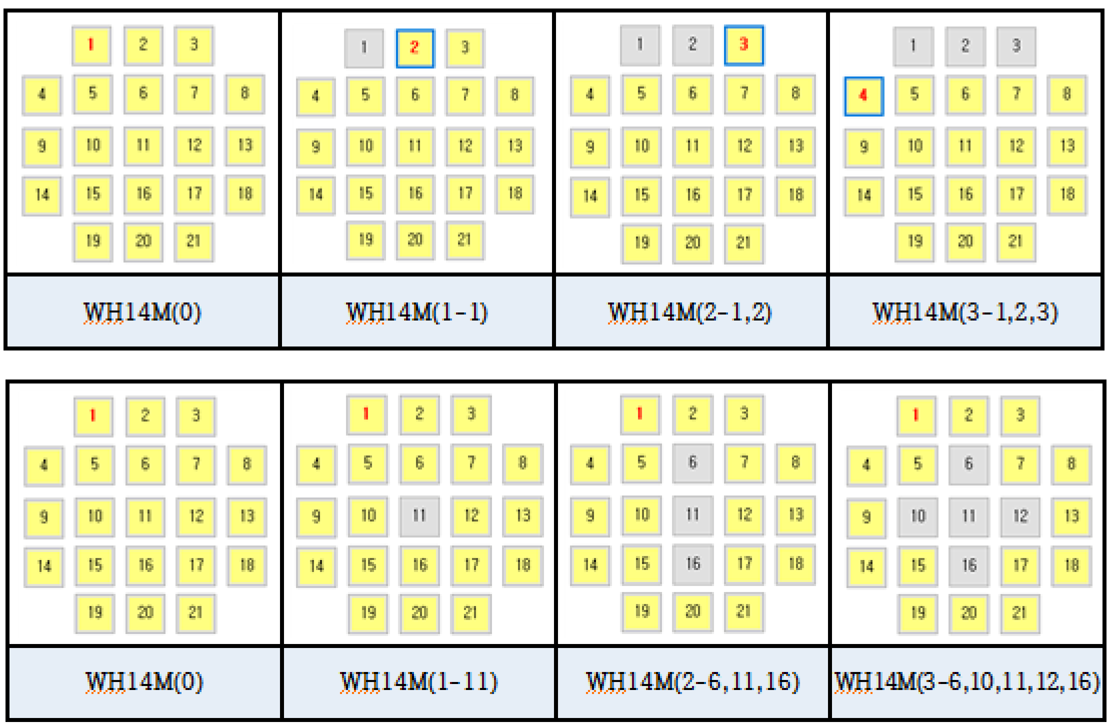

Fig. 4. The assumed sabotage events, missing some assemblies from side or center of a cask

Since enrichment and burnup rate range of the nuclear fuel generally used in Korea was 2~5 wt % and 25,000~50,000 MWD/MTU, the input parameters respectively set up 4.5 wt %, 45,000 MWD/MTU, 10 years, and WH 14x14 assemblies for the initial enrichment rate, burn-up rate, cooling time, and fuel type. Besides, neutron flux on the side of a cask was considered for this sabotage events due to an experimental concept shown as Fig. 2. The changes of the neutron flux for each sabotage events are represented Fig. 5 and 6.

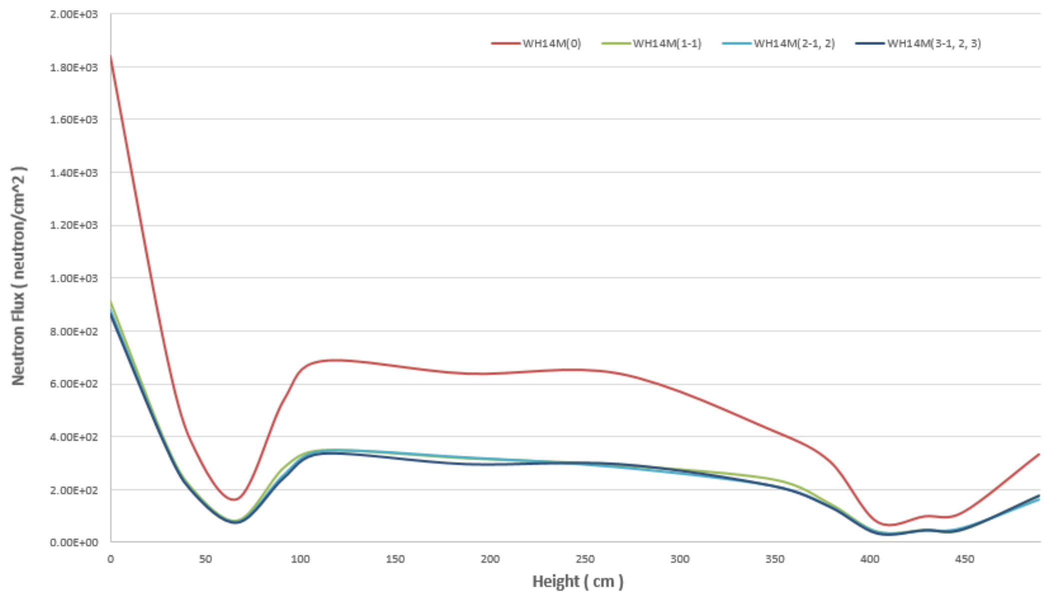

Fig. 5. The changes of the neutron flux for the case missing some assemblies from the side of a cask

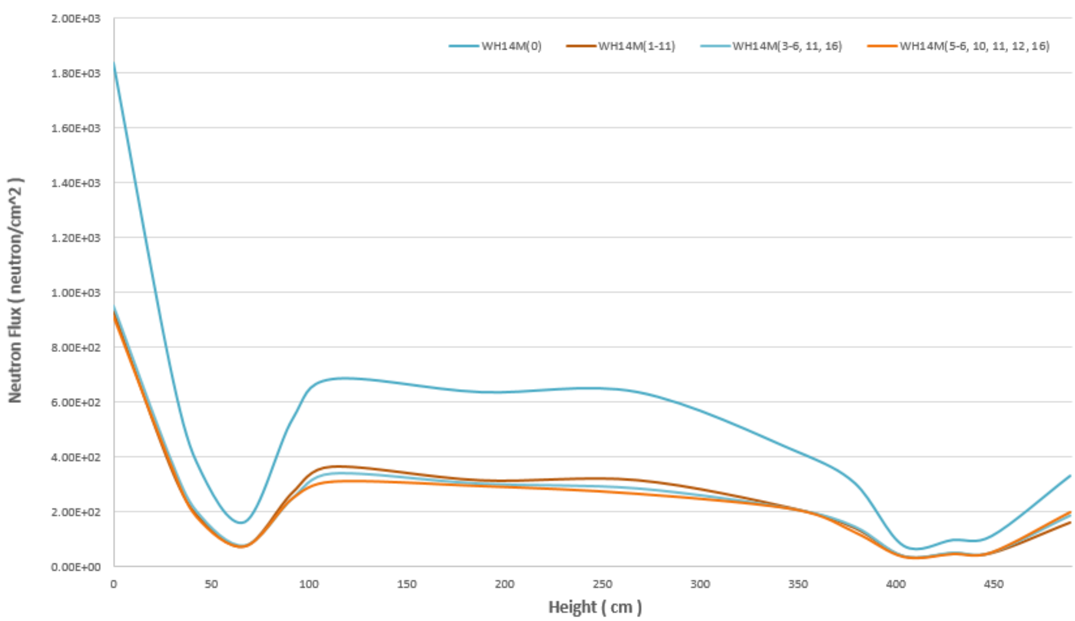

Fig. 6 . The changes of the neutron flux for the case missing some assemblies from the center of a cask

As shown in Fig. 5 and 6, neuron flux was significantly reduced when an assembly was missed. However, it suffers to clearly define the number of missing assemblies.

CONCLUSION

This study has laid the groundwork for a new experimental approach, a new spent nuclear fuel monitoring system employing He-4 gas scintillation fast neutron detectors. A new algorithm and user-interface program, coupled ORIGEN-ARP and MCNP, makes a user to easily evaluate neutron flux on the surface of a cask.

In the study for sabotage events, this neutron monitoring system can declare whether some sabotage events occur or not. However, it suffers to define the number of missing assemblies. Further steps will be to (1) update the developed program by applying various types of casks, (2) study for other cases of sabotage events, and (3) try to intercompare the simulation outputs with the experimental results.

REFERENCES

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[2] CCC-0371 ORIGEN-2.2. https://www.oecd-nea.org/tools/abstract/detail/ccc-0371

[3] J. Ko and et al., "Shielding Analysis of Dual Purpose Casks for Spent Nuclear Fuel Under Normal Storage Conditions", Nuclear Engineering and Technology, Vol.46 No.4, 2014.

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