Evaluation of Radiation Shielding Performance of Disposal Canister Storing PWR Spent Fuels

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1. Introduction

Radiation shielding is an important factor in designing disposal canister containing spent nuclear fuel(SNF), because intensity for photon and neutron in SNF assembly after 40 year cooling is still high, ~10^{15} photons/sec and ~10^{8} neutrons/sec, respectively. Radiation escaping from the disposal canister emplaced in repository may cause radiolysis and form oxidizing chemical species. This may result in corrosion of canister itself to proceed. Personnel exposure is also important concern. If shielding performance of canister can reduce radiation level to 1mRem/hr, human access without a control on duration and frequency of exposure may be possible. This provides the benefit of more direct human control of waste packages handling and emplacement operations.

In this paper, the radiation shielding performance was evaluated based on current reference disposal system to check absorbed dose for radiolysis, and exposure dose for radiation protection.

2. Analysis Model and Computational Tool

2.1. Analysis Model

17×17 KOFA(Korea Optimized Fuel Assemblies) was chosen as the reference PWR SNF. Two types of enrichment were considered; 1) initial enrichment of 4.0wt% with discharged burnup of 45GWD/MTU as a representative of SNF currently stored, 2) initial enrichment of 4.5wt% with discharged burnup of 55GWD/MTU as a representative of high-bunup SNF to be produced in the near future.

Reference disposal system considered is as follows; A canister is filled with cast insert which contains four PWR fuel assemblies. Outer-shell of canister is made of corron resistant material such as Ni alloy or Cu. In this study, 5cm thick Ni alloy was considered for outer-shell of the canister. The outer diameter of canister is 119cm. The canister is surrounded by 50cm thick bentonite buffer. Dimensions and basic data related to reference SNF and disposal system are described in detail in Fig. 1 and reference [1].

2.2. Computational Tool and Assumptions

ORIGEN2 and MCNP[2] were applied in the radiation shielding analysis. ORIGEN2 code was undertaken to generate neutron and photon intensity of SNF. This calculation provided initial conditions for the following analysis, in which MCNP were done to determine the dose rates. In the MCNP calculation, MCNPLIB02 library for photon and neutron cross section library generated from ENDF-VI were used.

For absorbed dose rate in bentonite buffer, it was assumed that the buffer was fully saturated with water. And then, the only energy transferred to water in bentonite buffer was considered. For exposure dose rate, flux to dose conversion factor proposed by ICRP74 was applied after neutron or photon flux at desired position was estimated.

Spontaneous fission neutron source, (α, n) source, and secondary gamma rays produced by neutron were considered to evaluate dose rate for neutron. A flat source term distribution in the axial and radial direction was assumed. The source intensity and spectra needed for the analysis is also described in detail in reference [1].

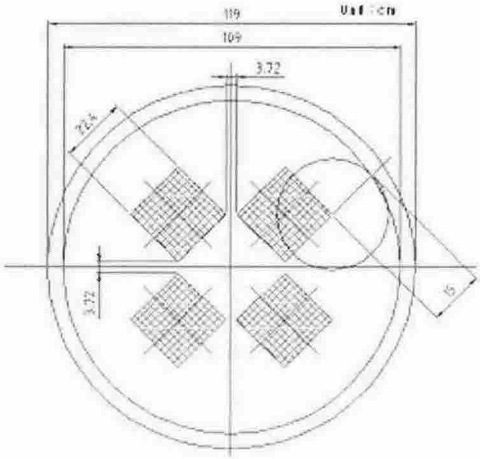

Fig. 1. Configuration and Dimensions of the Canister

3. Results

3.1. Absorbed Dose

Absorbed dose rate from the surface of canister should be maintained below the limit of ~1Gy/hr to avoid radiolysis and subsequent corrosion previously mentioned. Fig. 2 represents the absorbed dose rate in the water in bentonite layer close to the canister surface. When the absorbed dose rate was calculated, the 5cm thick Ni alloy surrounding the cast insert was preserved corresponding to increase in diameter of cast insert.

As shown in Fig. 2, if the outer radius of cast insert is beyond ~42cm, the absorbed dose rate is maintained under the limit of 1 Gy/hr. This value corresponds to 4cm thick carbon steel. The radiation shielding performance for current reference canister design was also indicated in Fig. 2. From the figure, it can be concluded that current design sufficiently satisfies the design limit, and that radiolysis induced by neutron and photon is not an important factor in further optimization of the disposal canister.

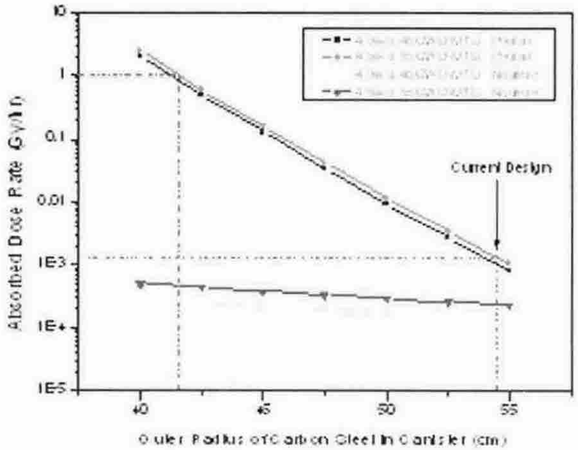

Fig. 2. Absorbed Dose Rate as a function of Cast Insert Thickness for Neutron and Photon

3.2. Exposure Dose

As lower surface dose rate is achieved, the more benefits regarding radiation protection in the emplacement operation of waste package are possible. As the shield thickness increases to reduce the dose rate, however, cost and weight are increased. Therefore, from the aspect of safety and cost point of view, dose rate at the surface of the canister should be weighed. As a result, the dose rates at the surface of canister currently proposed were shown to be 0.19 and 0.14 Rem/hr for photon and neutron, respectively. This means that it is impossible for worker in repository to access the canister during emplacement without control on duration and frequency of exposure. Fig. 3 and Fig. 4 represent the dose rate distribution along the distance from the center of canister. These results indicate that contribution of photon and neutron to personnel exposure is nearly same, and photon dose rate depend on the arrangement of SNF embedded in cast insert. When the distance from the canister is farther than 10m, the dose rate is reduced to 1/100 of the value at the surface of canister. To allow the personnel to access the canister without control, additional shielding mechanism should be developed.

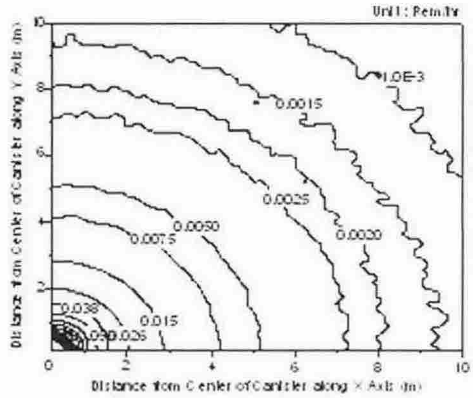

Fig. 3. Dose Rate Distribution for Neutron (4.0w/o)

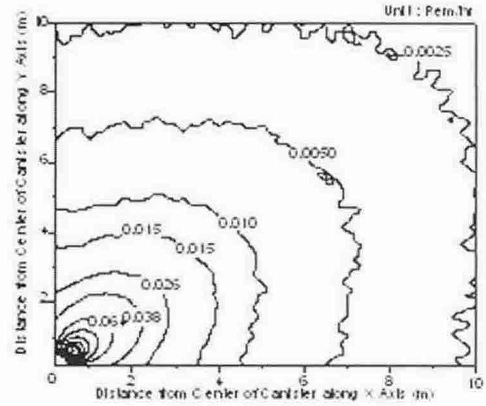

Fig. 4. Dose Rate Distribution for Photon (4.0w/o)

4. Conclusions

Absorbed dose and exposure dose were estimated on the basis of current disposal system containing reference SNF. It was shown that radiolysis associated with corrosion was not an important factor in the design of disposal canister, and that dose rate of 0.33 Rem/hr was expected at the surface of the canister, which means additional shielding mechanism should be developed to allow worker to access the waste package without control in emplacement operation.

ACKNOWLEDGEMENTS

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REFERENCES

[1] J. W. Choi, et. al, "Reference Spent Fuel and Its Characteristics for the Concept Development of a Deep Geological Disposal System," KAERI/TR-914/97, Korea Atomic Energy Research Institute, 1997.

[2] J. F. Briesmeister, "MCNP-A General Monte Carlo N-Particle Transport Code, Version 4C," LA-13709-M, Los Alamos National Laboratory, 2000.