Design of High Density Spent Fuel Storage Rack Applying Burnup Credit

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1. Introduction

The spent fuel from nuclear power plants is stored in a water tank in the power plant. It is expected that the temporary storage facilities in these nuclear power plants will be saturated by 2022. The intermediate storage facility in Korea will be completed by 2035. Hence, according to the spent fuel management plan, it is necessary to secure additional temporary storage facilities on the site by that time. As an alternative, storage capacity can be increased by replacing or adding re-racking of existing spent fuel storage to high density spent fuel storage. The high density spent fuel storage means reducing the space between fuel assemblies and allows more spent fuel to be stored. However, as more spent fuel is inserted into a space of equal size, the criticality increases. The criticality safety of spent nuclear fuel should be maintained at the same level. In this study, placing the annular cylinder type of neutron absorber into the water holes of assemblies and changing the type and concentration of the material used in the neutron absorber is proposed to reduce the spacing between fuel assemblies.

Until now, spent nuclear fuel has been evaluated conservatively by assuming spent nuclear fuel as a fresh fuel in most nuclear fuel criticality safety analysis. However, the spent nuclear fuel contains fission products with a large neutron absorption cross section. Therefore, the criticality should be evaluated in consideration of this fact. Hence, the burnup credit is applied considering the combustion of spent fuel. Criticality calculations are performed using two continuous energy Monte Carlo neutron transport codes MCNP6 [1] and MCS [2] developed by the COmputational Reactor physics and Experiment laboratory (CORE) in Ulsan National Institute of Science and Technology (UNIST) with the nuclear cross-sectional data library, ENDF/B-VII.0 [3]

2. Calculation Model

2.1. Fuel assembly

The calculational model is the assemblies used in APR-1400 reactor being used at Shin Kori Unit 3. The fuel assembly has 236 fuel rods, five guide tubes in a 16 x 16 array. The geometry specifications are illustrated in Fig 1.

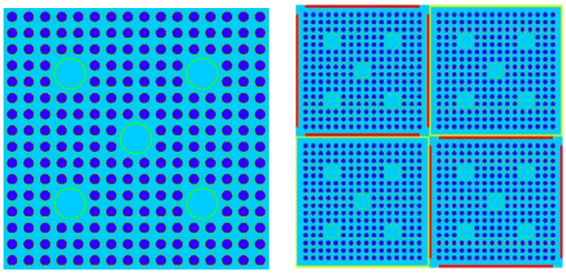

Fig. 1. The fuel assembly (left) and region II (right)

2.2. Spent fuel storage rack

The spent fuel pool is made of region I and region II. The region I rack can store both irradiated and unirradiated fuel. However, the region II is a storage for only irradiated fuel assemblies. Therefore, the burnup credit is widely accepted to establish effective criticality evaluation for region II [4]. The conventional design (CD) of Region II is described in Fig 1. Criticality calculations are performed using the in-house Monte Carlo code, MCS with the nuclear cross-sectional data library, ENDF/B-VII.0 and the results are described in Table I. The burnup credit calculation is also conducted using MCS code. The spent fuel of 1.72 wt% initial enrichment is depleted with the average burnup of 40,000 MWD/MTU using 10 axial burnable zones [5]. The isotopic number densities from depletion calculations using MCS code are used in criticality calculations of region II.

Table I: The neutron multiplication factor of region II

	k_{eff}
Fresh fuel(MCNP)	0.90339 ± 0.00010
Fresh fuel(MCS)	0.90346 ± 0.00016
Depleted Fuel(MCS)	0.90181 ± 0.00006

3. Sensitivity Test

3.1. Thickness of neutron absorber

The conventional neutron absorber consists of single plate between the assemblies, which increases the rack pitch. The shape of the neutron absorber presented in this study is an annular cylinder form that can be inserted into the guide tube position. The spent fuel storage facilities must be maintained in a subcritical state, and thus the neutron multiplication factor (k_{eff}) should be lower than that of conventional design. Fig 2.

shows the neutron multiplication factor with the thickness of the neuron absorber in the annular cylinder. The thickness of 0.643 cm maintains the lowest neutron multiplication factor value of 0.94841.

As the thickness of the neutron absorber increases, the amount of neutron absorber increases. However, the multiplication factor is not proportional to the change in the thickness of the neutron absorber. For example, as shown in Fig. 2, the multiplication factor is decreased to neutron thickness of about 0.6 cm. From the neutron thickness of about 0.6 cm, the multiplication factor increases as the thickness increases. This phenomenon can be explained by the correlation between the neutron absorber thickness and the flux trap. That is, flux trap can be decreased by increasing the neutron absorber thickness since moderation of the fast neutrons is decreased as the quantity of water as moderator.

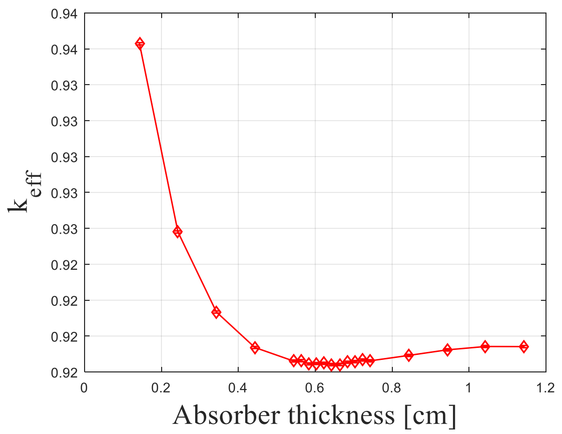

Fig. 2. The k_{eff} with the annular cylinder type neutron absorber thickness.

3.2. Material of neutron absorber

The most commonly used neutron absorber material s are gadolinium and boron since they have high neutron absorption cross section. Gd-157 isotopes have about 60 times higher neutron absorption cross section compared to B-10. Gadolinium is the base material for neutron absorber and the other candidates are B-10, Er-167, Eu-151 and Sm-149 due to their high thermal neutron absorption cross section. The Fig. 3 shows the neutron absorption cross section for those candidates of absorber material.

The neutron multiplication factor is calculated with applying the various material types and concentration. The additional absorber material use with gadolinium can compensate the neutron absorption in energy region that Gd-157 has low neutron absorption cross section and it is effective in lowering multiplication factor according to the result of Fig. 4. Among the candidate material of B-10, Er-167, Eu-151 and Sm-149, Eu-167 shows good performance as an effective material of annular cylinder type neutron absorber.

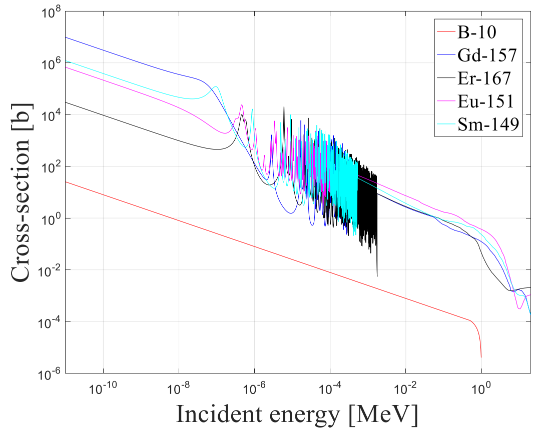

Fig. 3. The neutron absorption cross section of selected candidates for neutron absorber material.

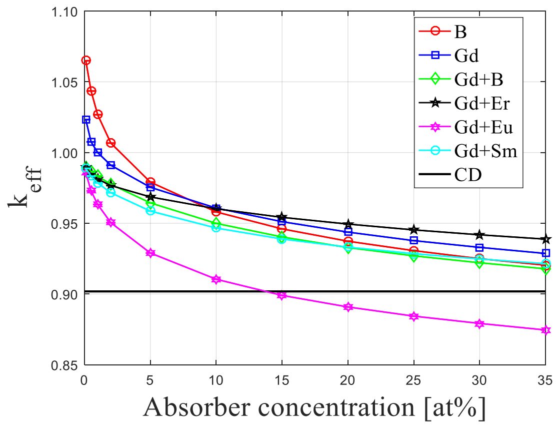

Fig. 4. The comparison of neutron absorption capacity of various materials.

3.3. Rack Pitch

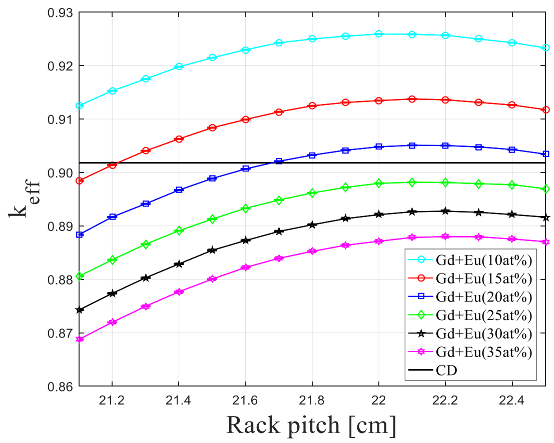

Fig. 5. Optimization of absorber concentration with the reduced rack pitch.

The rack pitch can be reduced using an annular cylinder type neutron absorber and a neutron absorber composed of Gd and Eu. The rack pitch can be reduced from 22.6 cm to 21.1 cm, under the condition that the neutron multiplication factor is lower than that of conventional design. The optimum concentration of neutron absorber that meets this criterion is Gd 2.0 and Eu 15.0 atomic percent.

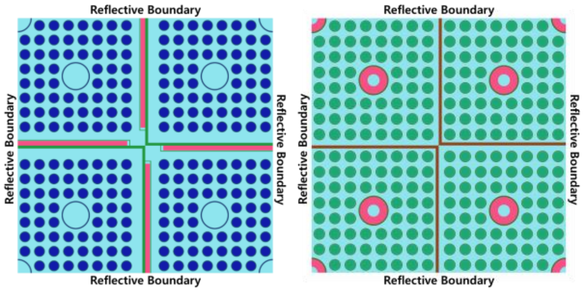

Fig. 6. The conventional (left) and proposed (right) design of region II.

4. Conclusion

This study proposes a high density spent fuel storage rack design for region II as a plan to increase the storage capacity of the temporary storage facilities in the nuclear power plants. The depletion calculations and criticality evaluation analysis are performed using two continuous energy Monte Carlo neutron transport codes MCNP6 and MCS. Analysis model is the APR-1400 and PLUS7 fuel models of the Shin-Kori nuclear power plant. As results of design of high density spent fuel storage rack, the spent fuel storage capacity can be increased by 14.7%.

ACKNOWLEDGEMENT

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