Transactions of the Korean Nuclear Society Autumn Meeting
Yeosu, Korea, October 25-26, 2018

Assessment on PWR Fuel Depletion and Neutron Multiplication Factors for High Density Spent Fuel Pool

Sanggeol Jeong, Wonkyeong Kim, Matthieu Lemaire, Deokjung Lee*
Department of Nuclear Engineering, Ulsan National Institute of Science and Technology,
50 UNIST-gil, Ulsan, 44919, Republic of Korea
**Corresponding author: deokjung@unist.ac.kr*

1. Introduction

The composition of nuclear fuel changes during depletion in nuclear reactors. As a result, compared to fresh fuel, spent nuclear fuel contains actinides and fission products with large neutron absorption cross-sections and anti-reactivity effect. However, in most nuclear criticality safety evaluation, the spent nuclear fuel is evaluated conservatively by modelling the spent nuclear fuel as fresh fuel. This assumption is a very conservative safety margin and increases the spent nuclear fuel storage cost. Taking credit for the reduced reactivity of spent nuclear fuel in criticality safety analyses on spent nuclear fuel handling facilities is referred to as burnup credit. In this study, the design of high density spent fuel storage rack is proposed by using annular cylinder neutron absorbers and by applying burnup credit. Through the installation of dense racks, the spent fuel storage capacity can be increased.

Criticality calculations are performed using two continuous energy Monte Carlo neutron transport codes MCNP6 [1] and MCS [2], the latter being developed by the COmputational Reactor physics and Experiment laboratory (CORE) in Ulsan National Institute of Science and Technology (UNIST). The calculations are performed with the nuclear data library ENDF/B-VII.1 [3]

2. Calculation Model

2.1. Fuel assembly

A fuel assembly of APR-1400 reactor is chosen for depletion calculation using MCS. The selected APR-1400 reactor fuel assembly is composed of a 16 x 16 array of fuel rods, guide tubes and instrumentation tube. Each fuel assembly has 236 fuel pins as shown in Fig. 1. The fuel material is uranium dioxide (UO2) surrounded by thin-walled zircaloy cladding. The fuel pin pitch is 1.285 cm and the outer radius of cladding is 0.475 cm. The maximum initial enrichment of the fuel is 4.50 w/o ^{235}U. The fuel assembly is depleted up to 60 MWd/kgU burnup.

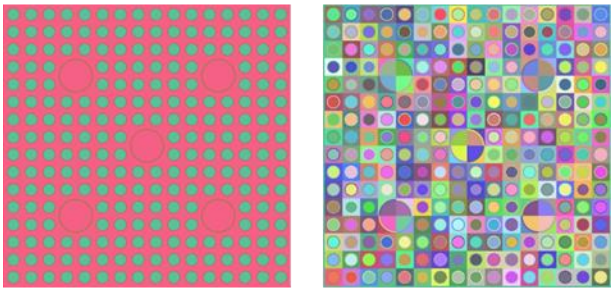


Fig. 1. Top view of APR-1400 fuel assembly depletion calculation model.

2.2. Spent fuel pool

The main function of spent fuel storage facilities is to safely store the fresh and spent fuel assemblies in a water tank. The spent fuel pool is usually divided into two regions. As shown in Fig. 2, one of the regions is region I designed for the storage of fresh (not irradiated) fuel. The other region is region II and is used for the storage of irradiated fuel. Burnup credit is a widely accepted method to establish effective criticality evaluation for region II [4].

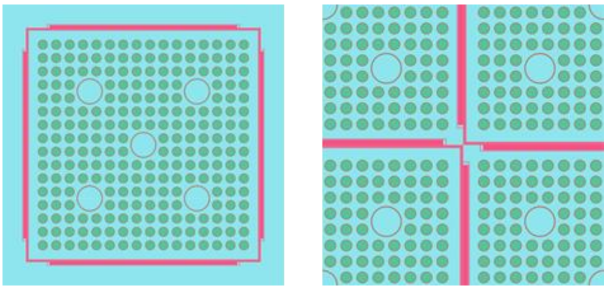


Fig. 2. Geometries of region I (left) and region II (right).

3. Criticality Analysis

3.1. Calculation information

The criticality results of MCNP6 and MCS for the conventional design of Region I and II are described in Table I. The fuel assembly depletion calculation is also conducted with the MCS code. Fig. 3 shows the neutron multiplication factors calculated with MCS for infinite fuel assembly problem. The 16 initial enrichments from 1.72 wt% to 4.50 wt% ^{235}U are depleted with the average burnup of 60 MWD/kgU using the fuel pins divided into 18 axial burnable zones [5]. The isotopic number densities from the MCS depletion calculation are then used in criticality calculations of region II to simulate the spent fuel with burnup credit.

In comparison to the fresh fuel, the spent fuel contains nuclides that lead to a change in the reactivity. The nuclides considered for burnup credit are divided into three groups based on the importance to fuel reactivity. As shown in Table II, twelve actinide and sixteen fission product isotopes are selected from the reference document NUREG/CR-7109 [6] of the Nuclear Regulatory Commission (NRC). Fig. 4 shows the multiplication factors calculated by MCS for the regions I and II with fuel at 3.14 wt% initial enrichment.

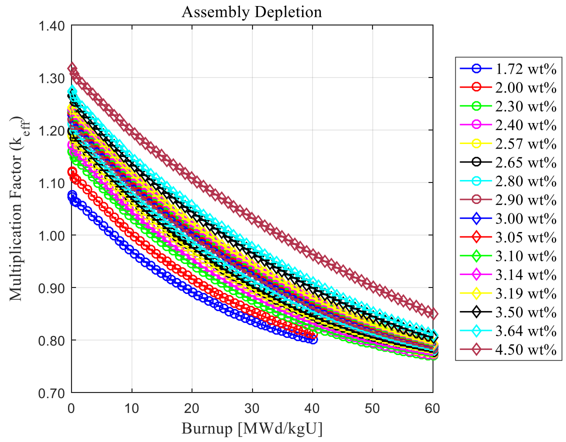


Fig. 3. Neutron multiplication factor for different initial enrichment as function of burnup.

Table I. Neutron multiplication factor of conventional-design region I and II with 1.72 wt% initial enrichment

k_{eff}	Region I	Region II	1 σ (pcm)
Fresh fuel (MCNP6)	0.69620	0.86831	11
Fresh fuel (MCS)	0.69657	0.86881	13
Depleted fuel (MCS)	-	0.80975	12

Table II. Nuclides used in applying burnup credit criticality analysis

Set of nuclides for actinide-only burnup credit (12)			
^{234}U	^{235}U	^{236}U	^{238}U
^{237}Np	^{238}Pu	^{239}Pu	^{240}Pu
^{241}Pu	^{242}Pu	^{241}Am	^{243}Am
Set of nuclides for actinides and fission product (16)			
^{95}Mo	^{99}Tc	^{101}Ru	^{103}Rh
^{109}Ag	^{133}Cs	^{143}Nd	^{145}Nd
^{147}Sm	^{149}Sm	^{150}Sm	^{151}Sm
^{152}Sm	^{151}Eu	^{153}Eu	^{155}Gd

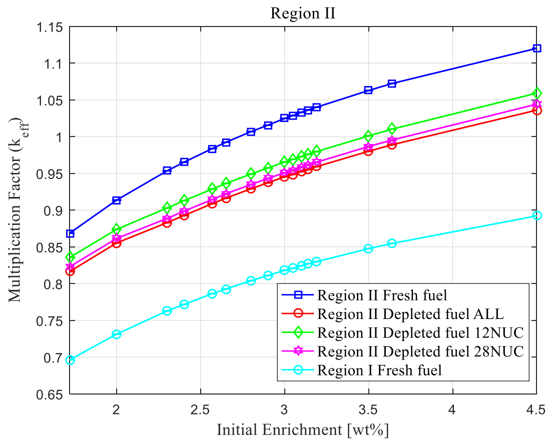


Fig. 4. Neutron multiplication factor for region I and II as function of initial enrichment. Error bars represent 1σ statistical uncertainties.

4. Sensitivity Analysis

4.1. Thickness of neutron absorber

The high-density rack is designed by inserting a neutron absorber in the form of an annular cylinder into a water hole in the fuel assembly. The radius of the fuel assembly water hole which is used for instrumentation during reactor operation is 1.1430 cm, which is therefore the maximum possible radius for the hollow-cylindrical neutron absorber. For region II, annular cylinder neutron absorbers are used instead of neutron absorber plates while both conventional absorber plates and annular cylinders made of neutron absorber are used together for region I. The proposed design of high density spent fuel pool is shown in Fig. 5. Fig. 6 shows the behavior of the neutron multiplication factor with respect to the change in the outer radius of the annular cylinder neutron absorber. Through the sensitivity analysis, the optimal radius of the annular cylinder to reduce the criticality was determined. Radius values of 0.823 cm and 0.603 cm are selected as optimum values in regions 1 and 2, respectively, and the criticality with these designs amounts to 0.64009 and 0.97631, respectively.

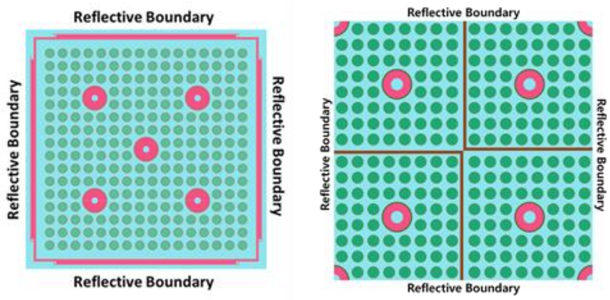


Fig. 5. Proposed design of region I (left) and region II (right).

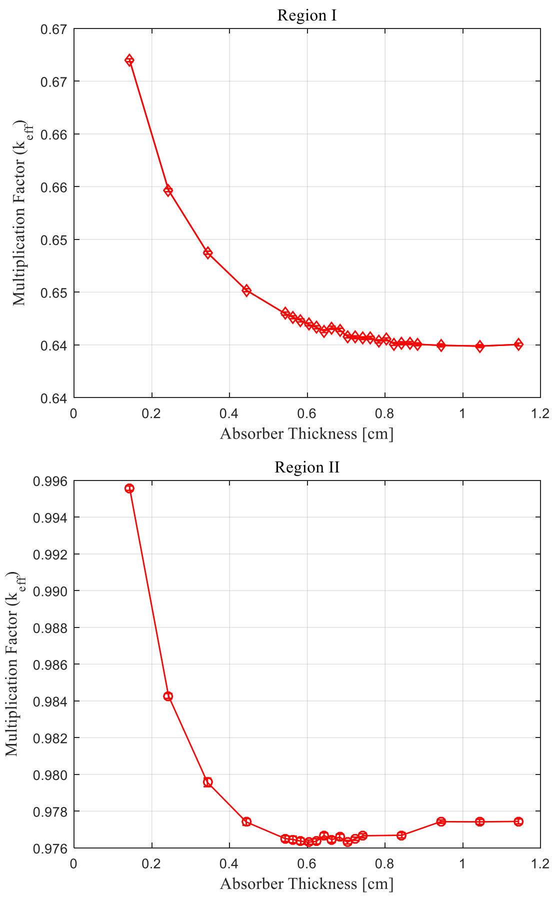


Fig. 6. k_{eff} as function of the radius of annular cylinder type of neutron absorber for region I (Above) and region II (Below).

4.2. Material of neutron absorber

Fig. 7 and 8 show the neutron multiplication factor as a function of the concentration of different neutron absorber materials. The most commonly used neutron absorber materials are gadolinium and boron, since both isotopes have high neutron absorption cross section. Gadolinium as neutron absorber material shows better efficiency in lowering the neutron multiplication factor than boron. As a result, the optimum concentration of gadolinium is selected to be 2.0 at%. However, when the concentration of the neutron absorber material is sufficiently high, the amount of neutron absorption of ^{10}B is higher than that of ^{157}Gd since the ^{10}B nuclide has lower reaction cross section than the ^{157}Gd nuclide in the thermal region, but the reaction cross section of ^{10}B is larger in the fast region.

Based on 2.0 at% gadolinium, the performance of four additional neutron absorber material candidates [7-9] is analyzed. The candidates for additional neutron absorber material are ^{10}B, ^{167}Er, ^{151}Eu and ^{149}Sm. As can be seen from Fig. 7 and 8, ^{157}Gd and ^{151}Eu neutron absorber is the most effective combination to decrease the neutron multiplication factor.

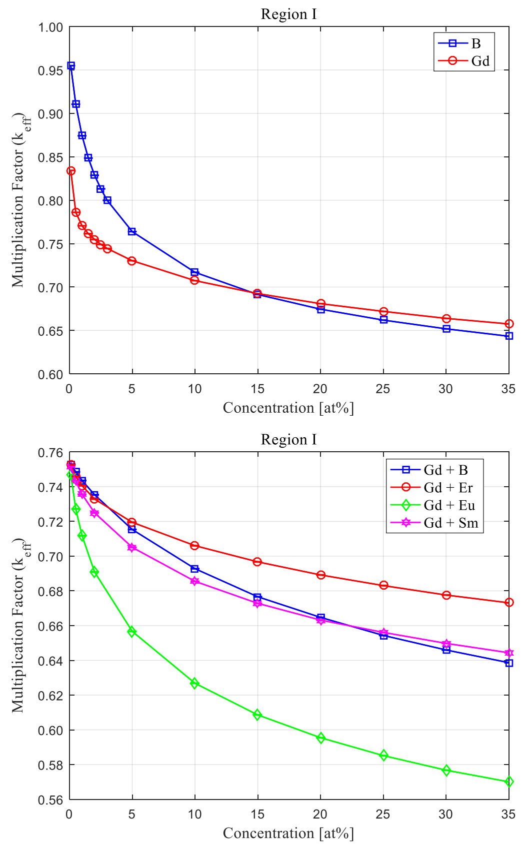


Fig. 7. Neutron multiplication factor as function of concentration of neutron absorber material candidates for region I.

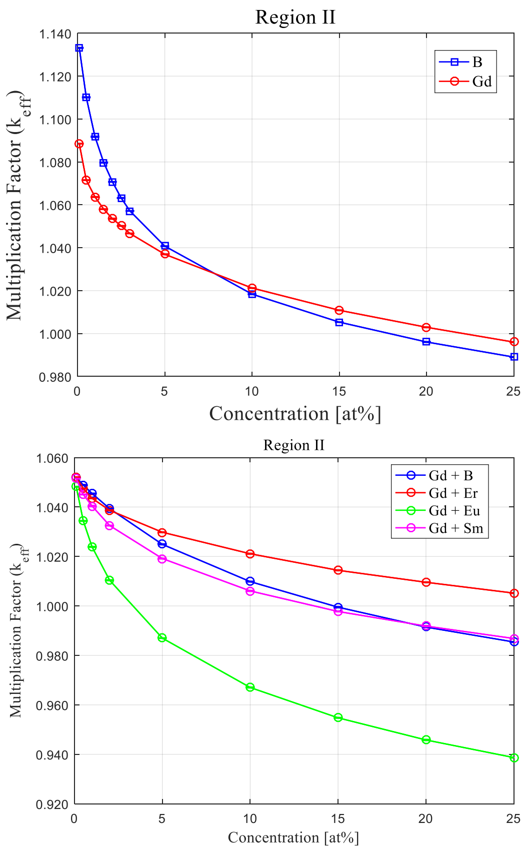
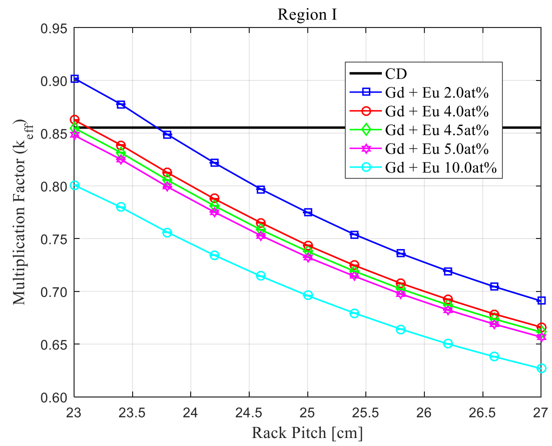


Fig. 8. Neutron multiplication factor as function of concentration of neutron absorber material candidates for region II.

4.3. Rack Pitch

Fig. 9 shows the neutron multiplication factor as a function of the rack pitch of region I and II when the selected optimum neutron absorber thickness is used. With a composition of Gd 2.0 + Eu 4.5 at%. the k_{eff} value equals 0.85487 for region I, which is less than the k_{eff} value of 0.85525 of the conventional design, for a dense rack with a pitch of only 23.00 cm compared to 27.00 cm in the conventional design. For the region II, the rack pitch is reduced from 22.60 cm to 21.10 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 4.5 at%.



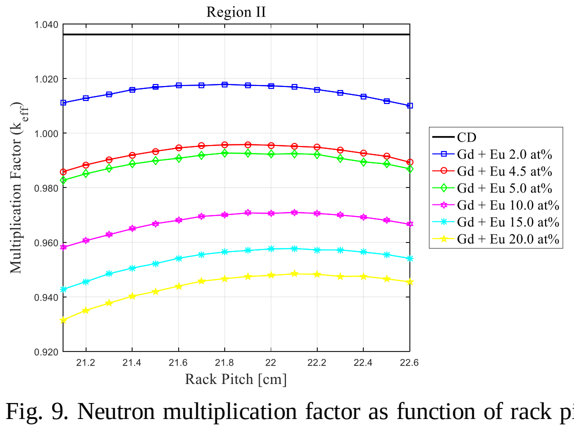


Fig. 9. Neutron multiplication factor as function of rack pitch for region I (Above) and region II (Below).

5. Conclusion

The research on a design to achieve a high density spent fuel storage rack is conducted. To satisfy the criticality safety requirements, the first strategy is to apply burnup credit for the nuclear fuel assemblies stored in the spent fuel storage and the second strategy is to employ neutron absorbers in the form of annular cylinder inserted into the guide tubes of the fuel assembly (in addition to conventional plate-type absorbers). The obtained results show that it is possible to reduce the rack pitch by 4.00 cm and 1.50 cm for the region I and II, respectively while achieving lower criticality level than in the conventional design of spent fuel pools.

ACKNOWLEDGEMENT

This research was supported by the project (L18S015000) by Korea Hydro & Nuclear Power Co. Ltd..

REFERENCES

[1] MCNP6 User's Manual. U.S.: Los Alamos National Laboratory; 2013, LA-CP-13-000634 Version 1.0.
[2] Hyunsuk Lee, Chidong Kong, and Deokjung Lee*, "Status of Monte Carlo Code Development at UNIST," PHYSOR2014, Kyoto, Japan, September 28 October 3 (2014)
[3] Chadwick M.B., "ENDF/B-VII.1 Nuclear Data for Science and Technology: Cross Sections, Covariances, Fission Product Yields and Decay Data," Nucl. Data. Sheet. 112:2887-2996, 2011.
[4] R.F. Mahmoud, M.K. Shaat, M.E. Nagy, S.A. Agamy, A.A. Abdelrahman, Burn-up credit in criticality safety of PWR spent fuel, Nuclear Engineering and Design 280 (2014) 628-633.
[5] U.S. NRC, "Recommendations for Addressing Axial Burnup in PWR Burnup Credit Analyses," 2013.
[6] J. M. Scaglione, "An Approach for Validating Actinide and Fission Product Burnup Credit Criticality Safety Analyses—Criticality (keff) Predictions," NUREG/CR-7109, ORNL/TM-2011/514, prepared for the U.S. Nuclear Regulatory Commission by Oak Ridge National Laboratory, Oak Ridge, Tennessee, Apr. 2012.
[7] M. Kim, H. Lee, J. Jong, D. Sohn, "사용후핵연료 저장 수조에서의 흡수체 물질별 핵적 성능 분석," KRS Autumn Meeting, Yeosu, Korea, Oct. 10-17. 2014.
[8] M. Kim, H. Lee, D. Sohn, "첨단흡수소재를 이용한 사용후핵연료 저장 수조에서의 핵적 성능 분석," KNS Spring Meeting, Jeju, Korea, May. 6-8. 2015.
[9] M. Kim, H. Lee, D, Sohn, "The effectiveness of Gd contents of fuel basket wall in UNF shipping & storage cask," Waste Management, Pheonix, AZ, USA, Mar 6-10. 2016.