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A Depletion Evaluation of PLUS7 16X16 Nuclear Fuel Assembly using TRITON

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1. Introduction

The capability of on-site storage of used nuclear fuels (UNFs) generated in the domestic nuclear power plants is project to reach the limit from 2024. So it is necessary to utilize a dry storage cask (DSC) for storage of UNFs and perform the accurate nuclear criticality safety evaluation of the DSC. A nuclear criticality safety evaluation that applies burnup credit (BUC) to a DSC is performed mainly through a two-step process: (1) the determination of isotopic compositions within UNFs to be loaded into a DSC by a depletion analysis and (2) the determination of the k_{eff} value with respect to the DSC by a criticality analysis [1]. In particular, isotopic compositions by a depletion analysis should be estimated accurately because the nuclides, especially major actinides such as uranium or plutonium, contained in a UNF has a significant influence on a depletion analysis and the subsequent criticality analysis. One way for an accurate estimation of isotopic compositions is to apply accurate cross section libraries to a depletion analysis. In this work, the new one-group cross section libraries of the ORIGEN code were generated with respect to a domestic nuclear fuel assembly (NFA), PLUS7, with and without gadolinia rods using the SCALE 6.1/TRITON code. Then, one of the new libraries was applied to the SCALE 6.1/STARBUCS code to evaluate the effect of the k_{eff} value.

2. Methods and Results

In this section, the SCALE 6.1/TRITON code used to model the PLUS7 16X16 was described, and the detailed modeling such as the design data and nuclides considered for depletion were specified. Finally, the thermal energy neutron flux distribution in burned NFA at the end of burnup and the fission and absorption cross-sections as functions of burnup were predicted. In addition, the effect of the k_{eff} value on the new PLUS7 16X16 cross-section library application was evaluated.

2.1 Computational Tool

TRITON (**T**ransport **R**igor **I**mplemented with **T**ime-dependent **O**peration for **N**eutronic depletion) is the very useful computational tool to estimate isotopic concentrations, source terms, and decay heat and create few-group homogenized cross sections for nodal core calculations. TRITON also can estimate the burnup of nuclear materials in configurations that have a strong spatial dependence on the neutron flux. TRITON uses a predictor-corrector approach to perform mainly a three-step process: (1) the cross-section processing sequence, (2) the transport sequence, and (3) the depletion sequence. At the end of a depletion calculation, new cross sections as a function of burnup in a format of ORIGEN can be obtained [2,3].

2.2 Main Input Settings

2.3.1 Design Data of NFA

The design of a domestic NFA was the PLUS7 16X16 at zero burnup, which consists of 236 fuel rods and five large guide tubes. For simplicity, the nuclear fuel assembly was a two-dimensional configuration. The design data for six types of NFA were listed in Table I. The values with prior to "/" corresponds to the fuel rods with normal enrichments in PLUS7 16X16 NFAs, while the values with posterior to "/" corresponds to the fuel rods with relatively lower enrichments. Fig. 1 shows the two-dimensional view of G0- and H0-type NFAs with 52 low-enriched fuel rods and without any gadolinia rod. Fig. 2 shows the two-dimensional view of G1- and H1-type NFAs with 52 low-enriched fuel rods and 8 gadolinia rods. Fig. 3 shows the two-dimensional view of G2- and H2-type NFAs with 52 low-enriched fuel rods and 12 gadolinia rods. Figs. 1-3 were printed by the SCALE 6.1/TRITON code.

Table I: Design data for six types of NFA

NFA Type	Fuel enrichment (w/o U^{235})	No. of fuel rods Per NFA	No. of Gd poison rods per NFA	Poison enrichment (w/o Gd_2O_3)
G0	4.10/3.62	184/52	0	0
G1	4.11/3.62	176/52	8	6.0
G2	4.12/3.61	172/52	12	6.0
H0	4.52/4.00	184/52	0	0
H1	4.50/4.00	176/52	8	6.0
H2	4.50/4.00	172/52	12	6.0

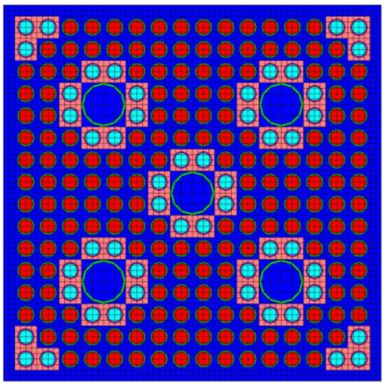


Fig. 1. Two-dimensional view of G0- and H0-type NFAs.

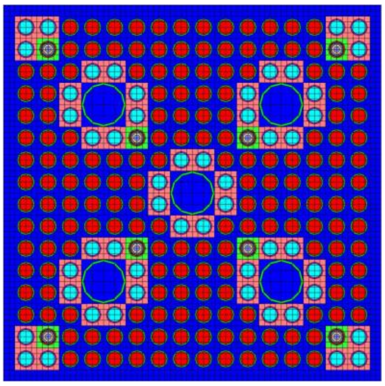


Fig. 2. Two-dimensional view of G1- and H1-type NFAs.

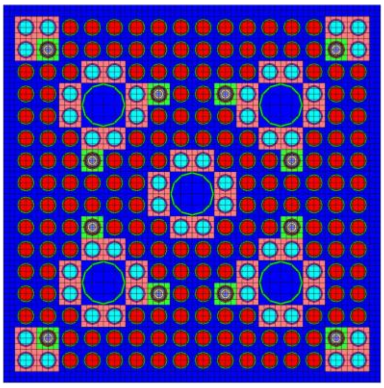


Fig. 3. Two-dimensional view of G2- and H2-type NFAs.

2.3.2 Nuclides Considered for Depletion

For depletion calculations of the SCALE 6.1/TRITON code, it is important to apply trace quantities of certain nuclides to the inventories of isotopic compositions to accurately track the nuclides' effect on cross-section processing and transport calculations as a function of burnup. 388 trace nuclides that could be selected the most were applied to the code, and the major nuclides of those were listed in Table II.

Table II: Two Nuclide sets [2]

Actinide nuclides				
Th-230	Th-232	Th-233	U-232	U-233
U-234	U-235	U-236	U-238	Np-237
Pu-238	Pu-239	Pu-240	Pu-241	Pu-242
Am-241	Am-242	Am-243	Cm-242	Cm-243
Cm-244	Cm-245	Cm-246	Bk-249	Bk-250
Cf-250	Cf-251	Cf-252	Cf-253	etc.
Fission products and Activation nuclides				
H-1	B-10	B-11	N-14	O-16
Kr-83	Nb-93	Zr-94	Mo-95	Tc-99
Rh-103	Rh-105	Ru-106	Ag-109	Sn-126
I-135	Xe-131	Xe-135	Cs-133	Cs-134
Cs-135	Cs-137	Pr-143	Ce-144	Nd-143
Nd-145	Nd-146	Nd-147	Pm147	Pm-148
Pm-149	Nd-148	Sm-147	Sm-149	Sm-150
Sm-151	Sm-152	Eu-151	Eu-153	Eu-154
Eu-155	Gd-152	Gd-154	Gd-155	Gd-156
Gd-157	Gd-158	Gd-160	Ta-182	etc.

2.4 Results and Evaluations

The SCALE 6.1/TRITON code was performed for the six types of PLUS7 16X16 NFA specified in previous subsection as a function of burnup. The burnup for the calculations ranged from 0 to 70,000 MWD/MTU. As a result, the following two results were obtained; neutron flux distributions and cross sections.

First, each of the 238 energy-group neutron flux distributions in the six types of PLUS7 16X16 NFA was calculated and plotted at the end of burnup. Because the number of energy group relevant to the thermal energy, 0.0253 eV, was 225, Fig. 4 indicated the representative flux distributions of neutron with thermal energy group number 225.

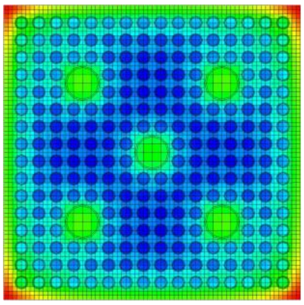
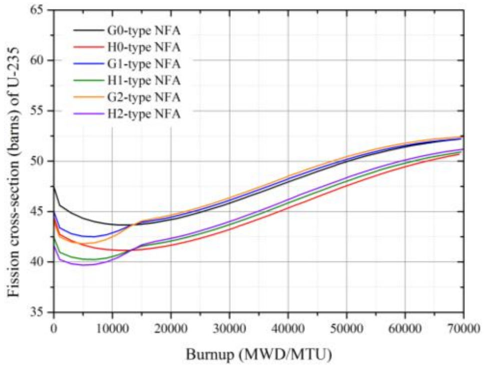


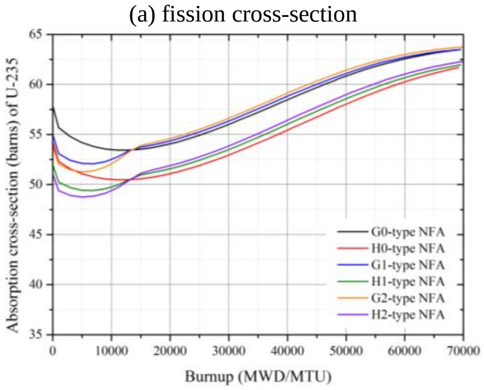
Fig. 4. Representative thermal neutron flux distribution.

Second, the fission and absorption cross-sections, in barns, of three major nuclides contained in the six types of PLUS7 16X16 NFAs were calculated and plotted as functions of burnup in MWD/MTU. Because the three major actinide nuclides that have a significant influence on a criticality analysis were uranium-235 and plutonium-239 as the fissile nuclides and uranium-238 as the largest amount of a nuclide, Figs. 5-7 indicated representatively the fission and absorption cross-section graphs of those nuclides. Fig. 5 shows the fission and absorption cross-sections with respect to U-235 in each type of NFA as a function of burnup. Fig. 6 shows the fission and absorption cross-sections with respect to Pu-239 in each type of NFA as a function of burnup. Fig. 7 shows the fission and absorption cross-sections with respect to U-238 in each type of NFA as a function of burnup. The black, red, blue, green, orange, and violet lines in

Figs. 5-7 express the G0-, H0-, G1-, H1-, G2-, and H2-type NFAs, respectively. The fission and absorption cross-sections of two fissile nuclides generated from burnups of low-enriched NFAs had higher values than those of high-enriched NFAs in Figs. 5-6, while the fission and absorption cross-sections of U-238 generated from burnups of low-enriched NFAs had lower values than those of high-enriched NFAs in Fig 7. Figs. 5-7 show that the fission and absorption cross-sections below approximately 15,000 MWD/MTU depended on the amount of gadolinia rods and the initial enrichment, while those above approximately 15,000 MWD/MTU depended on the only initial enrichment.

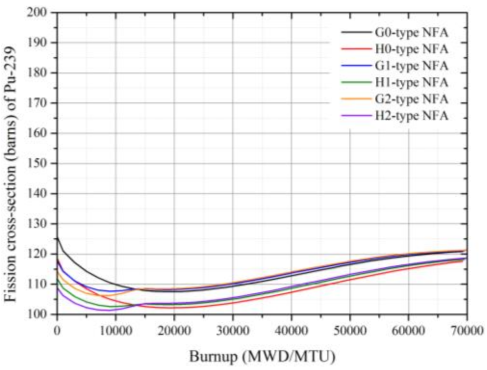


(a) fission cross-section

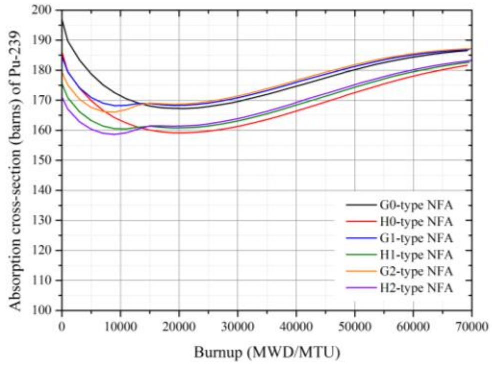


(b) absorption cross-section

Fig. 5. Cross-sections of (a) fission and (b) absorption, with respect to U-235 in each type of NFA, as a function of burnup.

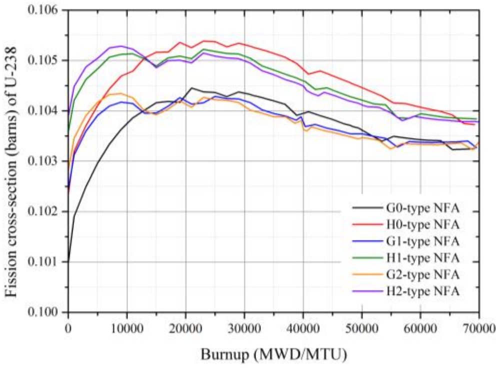


(a) fission cross-section

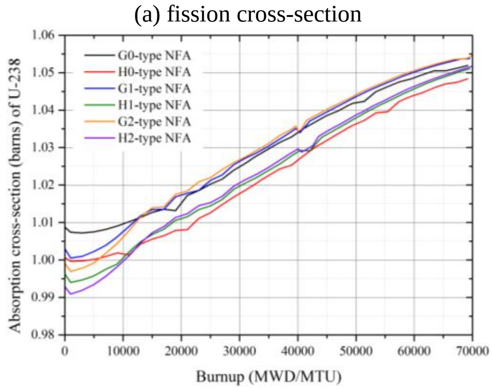


(b) absorption cross-section

Fig. 6. Cross-sections of (a) fission and (b) absorption, with respect to Pu-239 in each type of NFA, as a function of burnup.



(a) fission cross-section



(b) absorption cross-section

Fig. 7. Cross-sections of (a) fission and (b) absorption, with respect to U-238 in each type of NFA, as a function of burnup.

Finally, the k_{eff} values for the GBC-32 cask system stored in each of seven G2-type NFAs, of which the indices were 1, 11, 18, 20, 27, 38, and 49, in Ref. 4 were calculated using the new one-group cross section library for G2-type NFA by the SCALE 6.1/STARBUCS and MCNP6 codes. The results from this new library of G2-type NFAs with 12 gadolinia rods in this work were compared with the results from the CE 16X16 cross-section library without any gadolinia rod provided in the SCALE 6.1 code. The k_{eff} values for the 3 cooling times calculated by two codes are shown in Table 3. The k_{eff} values in round brackets are the results in Ref. 4, while the ones without round brackets result from the new library. The results show that the applications of the new library gave larger k_{eff}

values except for a few cases. The effects of the k_{eff} values on the new library ranged from -135 to 434 pcm in reactivity.

Table III: Value of thermal energy neutron flux

Cooling time	Index	STARBUCS		MCNP6	
		Uniform	Nonuniform	Uniform	Nonuniform
0 year	1	0.86817 (0.86491)	0.86360 (0.86179)	0.86919 (0.86646)	0.86478 (0.86362)
	11	0.87494 (0.87359)	0.87235 (0.87123)	0.87741 (0.87545)	0.87428 (0.87219)
	18	0.87510 (0.87256)	0.87151 (0.87050)	0.87740 (0.87493)	0.87409 (0.87208)
	20	0.90741 (0.90652)	0.90052 (0.90023)	0.90911 (0.90810)	0.90339 (0.90231)
	27	0.90800 (0.90655)	0.90255 (0.90148)	0.90952 (0.90775)	0.90445 (0.90339)
	38	0.92313 (0.92113)	0.91582 (0.91539)	0.92399 (0.92322)	0.91807 (0.91765)
	49	0.91120 (0.91052)	0.90538 (0.90359)	0.91359 (0.91275)	0.90675 (0.90562)
20 years	1	0.79634 (0.79426)	0.80410 (0.80352)	0.79757 (0.79573)	0.80461 (0.80463)
	11	0.80649 (0.80437)	0.81640 (0.81478)	0.80772 (0.80569)	0.81636 (0.81580)
	18	0.80627 (0.80406)	0.81528 (0.81468)	0.80833 (0.80572)	0.81620 (0.81564)
	20	0.84975 (0.84880)	0.85100 (0.85198)	0.85071 (0.85001)	0.85272 (0.85128)
	27	0.85024 (0.84873)	0.85365 (0.85267)	0.85068 (0.85024)	0.85547 (0.85389)
	38	0.86958 (0.86881)	0.86876 (0.86900)	0.87100 (0.87096)	0.87158 (0.87025)
	49	0.85517 (0.85373)	0.85483 (0.85513)	0.85624 (0.85568)	0.85742 (0.85595)
30 years	1	0.77902 (0.77791)	0.79090 (0.78918)	0.78040 (0.77826)	0.79205 (0.79030)
	11	0.78939 (0.78811)	0.80264 (0.80199)	0.79146 (0.78925)	0.80412 (0.80390)
	18	0.78989 (0.78771)	0.80337 (0.80326)	0.79119 (0.78931)	0.80397 (0.80334)
	20	0.83520 (0.83485)	0.84036 (0.84012)	0.83684 (0.83582)	0.84262 (0.84084)
	27	0.83667 (0.83503)	0.84412 (0.84122)	0.83743 (0.83603)	0.84420 (0.84476)
	38	0.85710 (0.85654)	0.86020 (0.85804)	0.85844 (0.85797)	0.86118 (0.86074)
	49	0.84108 (0.83985)	0.84539 (0.84506)	0.84233 (0.84212)	0.84574 (0.84594)

3. Conclusions

The thermal energy neutron flux distributions in burned NFA at the end of burnup and the fission and absorption cross-sections as a function of burnup were predicted for the PLUS7 16X16 NFAs using the SCALE 6.1/TRITON code. The results were calculated not only for the burnup range of 0 to 70,000 MWD/MTU, but also the initial enrichments and the amount of gadolinia rods in the six types of NFAs. In addition, the k_{eff} values for the GBC-32 cask system stored in each of seven G2-type NFAs were calculated using the new one-group cross section library for G2-type NFA by the SCALE 6.1/STARBUCS and MCNP6 codes. From the results calculated in these conditions, the following conclusions are drawn.

(1) The fission and absorption cross-sections of two fissile nuclides generated from burnups of low-enriched NFAs had higher values than those of high-enriched NFAs, while the fission and absorption cross-sections of U-238 generated from burnups of low-enriched NFAs had lower values than those of high-enriched NFAs.

(2) The fission and absorption cross-sections below approximately 15,000 MWD/MTU depended on the amount of gadolinia rods and the initial enrichment, while those above approximately 15,000 MWD/MTU depended on the only initial enrichment.

(3) The applications of the new PLUS7 16X16 cross-section library gave larger k_{eff} values than the CE 16X16 cross-section library, except for a few cases. The effects of the new library on the k_{eff} values ranged from -135 to 434 pcm in reactivity.

(4) The new one-group cross section libraries of a domestic PLUS7 16x16 NFA generated by SCALE 6.1/TRITON will be able to help the precise predictions of isotopic compositions with ORIGEN for the domestic PLUS7 NFAs.

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