Neutron Absorbing Ability Variation in Neutron Absorbing Material Caused by the Neutron Irradiation in Spent Fuel Storage Facility

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1. Introduction

In spent fuel storage facility like high density spent fuel storage racks and dry storage casks, spent fuels are stored with neutron absorbing materials installed as a part of those facilities, and they are used for absorbing neutrons emitted from spent fuels. Usually structural material with neutron absorbing material of racks and casks are located around spent fuels, so it is irradiated by neutrons for long time.

Neutron absorbing ability could be changed by the variation of nuclide composition in neutron absorbing material caused by the irradiation of neutrons[1]. After burnt in reactor, spent fuels are discharged and stored in spent fuel storage racks for some period and then they can be transported into spent fuel storage casks. So, neutron absorbing materials are continuously faced with spent fuels with boric acid solution or inert gas environment.

Major nuclides in neutron absorbing material are Al^{27}, C^{12}, B^{11}, B^{10} and they are changed to numerous other ones as radioactive decay or neutron absorption reaction. The B^{10} content in neutron absorbing material dominates the neutron absorbing ability, so, the variation of nuclide composition including the decrease of B^{10} content is the critical factor on neutron absorbing ability. In this study, neutron flux in spent fuel, the activation of neutron absorbing material and the variation of nuclide composition are calculated. And, the minimum neutron flux causing the decrease of B^{10} content is calculated in spent fuel storage facility. Finally, the variation of neutron multiplication factor is identified according to the one of B^{10} content in neutron absorbing material.

2. Methods and Results

2.1 Neutron Flux in Spent Fuel

It is necessary to calculate the usual neutron flux emitted from spent fuels with various burn-up conditions. For this calculation, PLUS7 fuel assembly with 16 x 16 array is applied. Neutron flux is added from two spent fuels because of the structure of spent fuel storage facility - one neutron absorbing material is installed between two spent fuels, and it is assumed that all neutron flux from two spent fuels directly irradiate neutron absorbing material simultaneously. For the conservative calculation, it is necessary to induce the maximum neutron flux with the condition described in Table 1.

Table 1. Spent Fuel Type and Burn-up Condition

Fuel Assembly Type	Number of Fuel Assembly	Cooling Time
PLUS7(16 x 16)	2	150 Hours
U^{235} Initial Enrichment and Burn-up		
2.0 wt%	2,500 MWD/MTU	
2.5 wt%	9,600 MWD/MTU	
3.0 wt%	16,000 MWD/MTU	
3.5 wt%	21,800 MWD/MTU	
4.0 wt%	28,500 MWD/MTU	
4.5 wt%	36,400 MWD/MTU	
5.0 wt%	42,700 MWD/MTU	

Fig. 1 is the typical cross sectional view of PLUS7 fuel assembly. It has 236 fuel rods, 4 guide thimble tubes and 1 instrumentation tube.

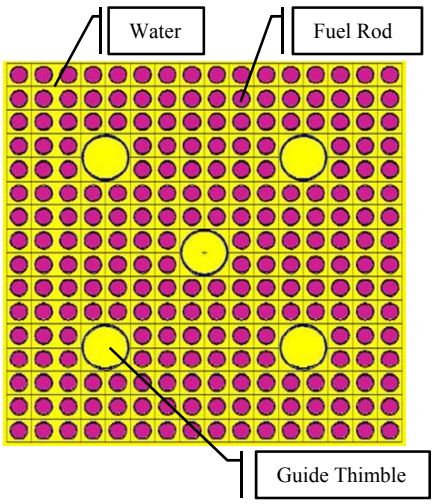

Fig. 1. Cross Sectional View of PLUS7 Fuel Assembly

Neutron flux is calculated and it is identified that as the U^{235} initial enrichment and burn-up are increased, neutron flux is increased accordingly. Therefore, the maximum neutron flux is produced on the condition of 5.0 wt% U^{235} - 42,700 MWD/MTU. Besides, as shown in Fig. 2, until 1 hour of cooling time after shut down in reactor, neutron flux is sharply decreased, and then by 150 hours, it is kept constantly.

Neutron flux in spent fuels is mostly composed of (α, n) reaction neutrons, spontaneous fission neutrons and delayed neutrons. Until 10 minutes of cooling time after shutdown, most neutron flux is composed of delayed neutrons with 10^{12} order, however, as time goes by, dominant neutron flux is composed of spontaneous fission neutrons and (α, n) reaction neutrons with 10^{8} order. Fig. 3 describes the neutron flux as per the neutron sources.

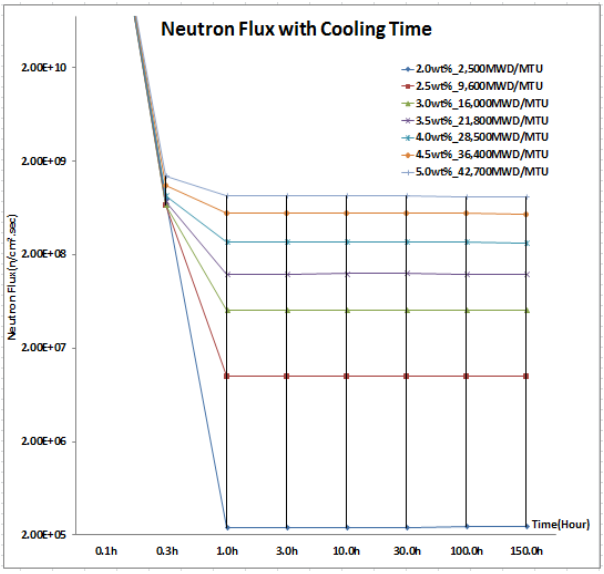

Fig. 2. Neutron Flux with Cooling Time(As per Burn-up Condition)

Around 150 hours of cooling time, over 90% of neutron flux is spontaneous fission neutrons and others are (α, n) reaction neutrons. Most actinides like Th^{230}, Th^{232}, Pa^{231}, U^{233}, U^{234}, U^{235}, U^{236}, U^{238}, Np^{127}, Pu^{236}, Pu^{238}, Pu^{239}, Pu^{240}, Pu^{241}, Pu^{242}, Pu^{244}, Am^{241}, Am^{243}, Cm^{242}, Cm^{243}, Cm^{244}, Cm^{245}, Cm^{246}, Cm^{248}, Cm^{250}, Bk^{249}, Cf^{249}, Cf^{250}, Cf^{252}, Cf^{254}, Es^{253}, Es^{254}, Es^{255} are the sources of spontaneous fission neutrons and U^{234}, Pu^{236}, Pu^{238}, Pu^{239}, Pu^{240}, Pu^{241}, Pu^{242}, Pu^{244}, Am^{241}, Am^{243}, Cm^{242}, Cm^{243}, Cm^{244} are the ones of (α, n) reaction neutrons.

Neutron flux and neutron absorbing material activation are calculated using ORIGEN-ARP[2] / ORIGEN-S[3] module in SCALE system developed by ORNL.

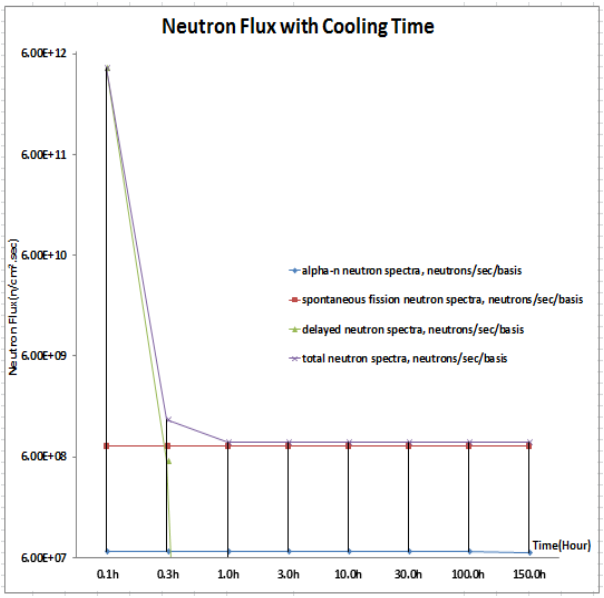

Fig.3. Neutron Flux with Cooling Time(As per Neutron Source)

2.2 Neutron Absorbing Material Activation

The major nuclides in aluminum based neutron absorbing material is Al^{27}, C^{12}, B^{10}, B^{11}. Table 2 shows the neutron absorbing material specification and nuclide composition before it is irradiated by neutrons.

Table 2. Neutron Absorbing Material Specification

Description	Dimension(cm)			Density (g/cc)	Weight Percent(%)			
	T	W	L		Al^{27}	C^{12}	B^{11}	B^{10}
Neutron Absorbing Material	0.42	19.4	381.0	2.713	72.01	6.08	17.87	4.04

Neutron absorbing material is irradiated by neutrons of 8.35E+08/cm^2-sec emitted from two fuel assemblies in spent fuel storage facility surrounded by water. It is assumed that neutrons directly irradiate the neutron absorbing material at least for 40 years continuously with the same neutron flux. Table 3 indicates the nuclide variation in neutron absorbing material after it is irradiated and there is little change on weight percent for Al^{27}, C^{12}, B^{10}, B^{11} . So, it is possible to expect that 10^8 level of neutron flux has little impact on the variation of nuclide composition in neutron absorbing material. It means that B^{10} content is little decreased caused by the neutron flux emitted from usual spent fuels.

Table 3. Nuclide Composition of Activated Neutron Absorbing Material

nuclide	Time (sec)							Fraction	%
	1.00E-04	3.00E-04	1.00E-03	3.00E-03	1.00E-02	3.00E-02	1.00E-01		
al27	6.07E+03	6.07E+03	6.07E+03	6.07E+03	6.07E+03	6.07E+03	6.07E+03	7.20E-01	72.02
b11	1.50E+03	1.50E+03	1.50E+03	1.50E+03	1.50E+03	1.50E+03	1.50E+03	1.78E-01	17.85
c12	5.06E+02	5.06E+02	5.06E+02	5.06E+02	5.06E+02	5.06E+02	5.06E+02	6.01E-02	6.01
b10	3.41E+02	3.41E+02	3.41E+02	3.41E+02	3.41E+02	3.41E+02	3.41E+02	4.05E-02	4.05
c13	6.15E+00	6.15E+00	6.15E+00	6.15E+00	6.15E+00	6.15E+00	6.15E+00	7.30E-04	0.07
li7	1.68E-01	1.68E-01	1.68E-01	1.68E-01	1.68E-01	1.68E-01	1.68E-01	1.99E-05	0.00
he4	9.60E-02	9.60E-02	9.60E-02	9.60E-02	9.60E-02	9.60E-02	9.60E-02	1.14E-05	0.00
si28	2.82E-04	2.82E-04	2.82E-04	2.82E-04	2.82E-04	2.82E-04	2.82E-04	3.34E-08	0.00
mg24	2.86E-06	2.86E-06	2.86E-06	2.86E-06	2.86E-06	2.86E-06	2.86E-06	3.39E-10	0.00
be10	2.84E-06	2.84E-06	2.84E-06	2.84E-06	2.84E-06	2.84E-06	2.84E-06	3.37E-10	0.00
h1	9.59E-07	9.59E-07	9.59E-07	9.59E-07	9.59E-07	9.59E-07	9.59E-07	1.14E-10	0.00
be9	7.39E-07	7.39E-07	7.39E-07	7.39E-07	7.39E-07	7.39E-07	7.39E-07	8.77E-11	0.00
h2	6.83E-08	6.83E-08	6.83E-08	6.83E-08	6.83E-08	6.83E-08	6.83E-08	8.11E-12	0.00
mg26	6.74E-08	6.74E-08	6.74E-08	6.74E-08	6.74E-08	6.74E-08	6.74E-08	8.01E-12	0.00
c14	2.75E-09	2.75E-09	2.75E-09	2.75E-09	2.75E-09	2.75E-09	2.75E-09	3.26E-13	0.00
he3	1.03E-09	1.03E-09	1.03E-09	1.03E-09	1.03E-09	1.03E-09	1.03E-09	1.22E-13	0.00
mg25	4.78E-10	4.78E-10	4.78E-10	4.78E-10	4.78E-10	4.78E-10	4.78E-10	5.67E-14	0.00
h3	4.40E-10	4.40E-10	4.40E-10	4.40E-10	4.40E-10	4.40E-10	4.40E-10	5.22E-14	0.00
na24	7.60E-11	7.60E-11	7.60E-11	7.60E-11	7.60E-11	7.60E-11	7.60E-11	9.02E-15	0.00
al28	1.88E-11	1.88E-11	1.88E-11	1.88E-11	1.88E-11	1.88E-11	1.88E-11	2.23E-15	0.00
n14	8.50E-12	8.50E-12	8.50E-12	8.50E-12	8.50E-12	8.50E-12	8.50E-12	1.01E-15	0.00
mg27	5.24E-12	5.24E-12	5.24E-12	5.24E-12	5.24E-12	5.24E-12	5.24E-12	6.22E-16	0.00
si29	3.69E-12	3.69E-12	3.69E-12	3.69E-12	3.69E-12	3.69E-12	3.69E-12	4.38E-16	0.00
li6	2.75E-12	2.75E-12	2.75E-12	2.75E-12	2.75E-12	2.75E-12	2.75E-12	3.26E-16	0.00
ne21	6.28E-15	6.28E-15	6.28E-15	6.28E-15	6.28E-15	6.28E-15	6.28E-15	7.46E-19	0.00
li8	2.30E-17	2.30E-17	2.30E-17	2.29E-17	2.28E-17	2.24E-17	2.12E-17	2.51E-21	0.00
be11	1.18E-17	1.18E-17	1.18E-17	1.18E-17	1.18E-17	1.18E-17	1.17E-17	1.39E-21	0.00
na23	5.68E-18	5.68E-18	5.68E-18	5.68E-18	5.68E-18	5.68E-18	5.68E-18	6.74E-22	0.00
ne22	3.03E-18	3.03E-18	3.03E-18	3.03E-18	3.03E-18	3.03E-18	3.03E-18	3.60E-22	0.00
b12	1.68E-17	1.67E-17	1.63E-17	1.52E-17	1.19E-17	6.01E-18	5.44E-19	6.46E-23	0.00
n15	4.04E-20	4.04E-20	4.04E-20	4.04E-20	4.04E-20	4.04E-20	4.04E-20	4.79E-24	0.00
si30	2.57E-20	2.57E-20	2.57E-20	2.57E-20	2.57E-20	2.57E-20	2.57E-20	3.05E-24	0.00
he6	3.17E-22	3.17E-22	3.17E-22	3.16E-22	3.14E-22	3.09E-22	2.91E-22	3.45E-26	0.00
be8	5.49E-23	5.49E-23	5.48E-23	5.48E-23	5.44E-23	5.35E-23	5.05E-23	6.00E-27	0.00
o18	3.61E-23	3.61E-23	3.61E-23	3.61E-23	3.61E-23	3.61E-23	3.61E-23	4.29E-27	0.00
mg28	1.64E-24	1.64E-24	1.64E-24	1.64E-24	1.64E-24	1.64E-24	1.64E-24	1.94E-28	0.00
ne20	6.15E-25	6.15E-25	6.15E-25	6.15E-25	6.15E-25	6.15E-25	6.15E-25	7.30E-29	0.00
ne23	2.06E-25	2.06E-25	2.06E-25	2.06E-25	2.06E-25	2.06E-25	2.06E-25	2.45E-29	0.00
na25	3.23E-26	3.23E-26	3.23E-26	3.23E-26	3.22E-26	3.22E-26	3.22E-26	3.82E-30	0.00
al29	2.55E-27	2.55E-27	2.55E-27	2.55E-27	2.55E-27	2.55E-27	2.55E-27	3.03E-31	0.00
Total							8.42E+03	1.00E+00	1.00E+02

The neutron spectrum produced from spent fuels is 0.4MeV ~ 6.4MeV for (α, n) reaction neutrons as shown in Table 4, and 0.1MeV ~ 6.4MeV for spontaneous fission neutrons, as shown in Table 5.

group	boundaries, MeV			initial	0.1h	0.3h	1.0h	3.0h	10.0h	30.0h	100.0h	150.0h	Fraction(%)
1	1.00E-11	-	1.00E-08	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00
2	1.00E-08	-	3.00E-08	2.61E-09	2.61E-09	2.61E-09	2.61E-09	2.61E-09	2.61E-09	2.61E-09	2.61E-09	2.61E-09	0.00
3	3.00E-08	-	5.00E-08	3.47E-09	3.47E-09	3.47E-09	3.47E-09	3.47E-09	3.47E-09	3.47E-09	3.47E-09	3.47E-09	0.00
4	5.00E-08	-	1.00E-07	8.67E-09	8.67E-09	8.67E-09	8.67E-09	8.67E-09	8.67E-09	8.67E-09	8.67E-09	8.67E-09	0.00
5	1.00E-07	-	2.25E-07	4.53E-05	4.53E-05	4.53E-05	4.53E-05	4.53E-05	4.53E-05	4.55E-05	4.57E-05	4.58E-05	0.00
6	2.25E-07	-	3.25E-07	6.35E-05	6.35E-05	6.35E-05	6.35E-05	6.35E-05	6.36E-05	6.37E-05	6.41E-05	6.42E-05	0.00
7	3.25E-07	-	4.00E-07	2.80E-04	2.80E-04	2.80E-04	2.80E-04	2.80E-04	2.80E-04	2.80E-04	2.78E-04	2.76E-04	0.00
8	4.00E-07	-	8.00E-07	3.93E-03	3.93E-03	3.93E-03	3.93E-03	3.93E-03	3.93E-03	3.93E-03	3.89E-03	3.86E-03	0.00
9	8.00E-07	-	1.00E-06	1.97E-03	1.97E-03	1.97E-03	1.97E-03	1.97E-03	1.97E-03	1.96E-03	1.95E-03	1.93E-03	0.00
10	1.00E-06	-	1.13E-06	1.28E-03	1.28E-03	1.28E-03	1.28E-03	1.28E-03	1.28E-03	1.28E-03	1.26E-03	1.25E-03	0.00
11	1.13E-06	-	1.30E-06	1.67E-03	1.67E-03	1.67E-03	1.67E-03	1.67E-03	1.67E-03	1.67E-03	1.65E-03	1.64E-03	0.00
12	1.30E-06	-	1.77E-06	4.62E-03	4.62E-03	4.62E-03	4.62E-03	4.62E-03	4.62E-03	4.62E-03	4.57E-03	4.53E-03	0.00
13	1.77E-06	-	3.05E-06	1.26E-02	1.26E-02	1.26E-02	1.26E-02	1.26E-02	1.26E-02	1.26E-02	1.25E-02	1.24E-02	0.00
14	3.05E-06	-	1.00E-05	6.84E-02	6.84E-02	6.84E-02	6.84E-02	6.84E-02	6.85E-02	6.84E-02	6.77E-02	6.72E-02	0.00
15	1.00E-05	-	3.00E-05	4.77E-01	4.77E-01	4.77E-01	4.77E-01	4.77E-01	4.78E-01	4.77E-01	4.73E-01	4.69E-01	0.00
16	3.00E-05	-	1.00E-04	7.27E+00	7.27E+00	7.27E+00	7.27E+00	7.27E+00	7.28E+00	7.27E+00	7.20E+00	7.15E+00	0.00
17	1.00E-04	-	5.50E-04	9.59E+01	9.59E+01	9.59E+01	9.59E+01	9.59E+01	9.59E+01	9.59E+01	9.50E+01	9.43E+01	0.00
18	5.50E-04	-	3.00E-03	1.03E+03	1.03E+03	1.03E+03	1.03E+03	1.03E+03	1.03E+03	1.03E+03	1.02E+03	1.01E+03	0.00
19	3.00E-03	-	1.70E-02	1.63E+04	1.63E+04	1.64E+04	1.64E+04	1.64E+04	1.64E+04	1.63E+04	1.62E+04	1.61E+04	0.00
20	1.70E-02	-	1.00E-01	2.51E+05	2.51E+05	2.51E+05	2.51E+05	2.51E+05	2.51E+05	2.51E+05	2.49E+05	2.47E+05	0.00
21	1.00E-01	-	4.00E-01	1.62E+06	1.62E+06	1.62E+06	1.62E+06	1.63E+06	1.63E+06	1.62E+06	1.61E+06	1.60E+06	0.02
22	4.00E-01	-	9.00E-01	3.40E+06	3.40E+06	3.40E+06	3.40E+06	3.40E+06	3.40E+06	3.40E+06	3.36E+06	3.34E+06	0.05
23	9.00E-01	-	1.40E+00	4.71E+06	4.71E+06	4.71E+06	4.71E+06	4.71E+06	4.72E+06	4.71E+06	4.67E+06	4.63E+06	0.07
24	1.40E+00	-	1.85E+00	6.73E+06	6.73E+06	6.73E+06	6.73E+06	6.73E+06	6.73E+06	6.72E+06	6.66E+06	6.61E+06	0.10
25	1.85E+00	-	3.00E+00	2.80E+07	2.80E+07	2.80E+07	2.80E+07	2.80E+07	2.80E+07	2.80E+07	2.77E+07	2.75E+07	0.41
26	3.00E+00	-	6.43E+00	2.42E+07	2.42E+07	2.42E+07	2.42E+07	2.42E+07	2.42E+07	2.41E+07	2.39E+07	2.37E+07	0.35
27	6.43E+00	-	2.00E+01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00
			total	6.89E+07	6.89E+07	6.89E+07	6.89E+07	6.89E+07	6.89E+07	6.88E+07	6.82E+07	6.77E+07	1.00

Table 4. (α, n) Neutron Spectrum from Spent Fuel

group	boundaries, MeV			initial	0.1h	0.3h	1.0h	3.0h	10.0h	30.0h	100.0h	150.0h	Fraction(%)
1	1.00E-11	-	1.00E-08	1.55E-03	1.55E-03	1.55E-03	1.55E-03	1.55E-03	1.55E-03	1.55E-03	1.53E-03	1.52E-03	0.00
2	1.00E-08	-	3.00E-08	2.18E-03	2.18E-03	2.18E-03	2.18E-03	2.18E-03	2.18E-03	2.18E-03	2.16E-03	2.15E-03	0.00
3	3.00E-08	-	5.00E-08	2.41E-03	2.41E-03	2.41E-03	2.41E-03	2.41E-03	2.41E-03	2.41E-03	2.40E-03	2.38E-03	0.00
4	5.00E-08	-	1.00E-07	7.20E-03	7.20E-03	7.20E-03	7.20E-03	7.20E-03	7.20E-03	7.20E-03	7.16E-03	7.13E-03	0.00
5	1.00E-07	-	2.25E-07	2.41E-02	2.41E-02	2.41E-02	2.41E-02	2.41E-02	2.41E-02	2.41E-02	2.40E-02	2.39E-02	0.00
6	2.25E-07	-	3.25E-07	2.43E-02	2.43E-02	2.43E-02	2.43E-02	2.43E-02	2.43E-02	2.43E-02	2.42E-02	2.41E-02	0.00
7	3.25E-07	-	4.00E-07	2.07E-02	2.07E-02	2.07E-02	2.07E-02	2.07E-02	2.07E-02	2.07E-02	2.06E-02	2.05E-02	0.00
8	4.00E-07	-	8.00E-07	1.39E-01	1.39E-01	1.39E-01	1.39E-01	1.39E-01	1.39E-01	1.39E-01	1.39E-01	1.38E-01	0.00
9	8.00E-07	-	1.00E-06	8.48E-02	8.48E-02	8.48E-02	8.48E-02	8.49E-02	8.49E-02	8.49E-02	8.45E-02	8.42E-02	0.00
10	1.00E-06	-	1.13E-06	5.99E-02	5.99E-02	5.99E-02	5.99E-02	5.99E-02	5.99E-02	5.99E-02	5.96E-02	5.94E-02	0.00
11	1.13E-06	-	1.30E-06	8.35E-02	8.35E-02	8.35E-02	8.35E-02	8.35E-02	8.35E-02	8.35E-02	8.32E-02	8.29E-02	0.00
12	1.30E-06	-	1.77E-06	2.59E-01	2.59E-01	2.59E-01	2.59E-01	2.59E-01	2.59E-01	2.59E-01	2.58E-01	2.57E-01	0.00
13	1.77E-06	-	3.05E-06	8.77E-01	8.77E-01	8.77E-01	8.77E-01	8.78E-01	8.78E-01	8.78E-01	8.74E-01	8.71E-01	0.00
14	3.05E-06	-	1.00E-05	7.73E+00	7.73E+00	7.73E+00	7.74E+00	7.74E+00	7.74E+00	7.74E+00	7.71E+00	7.68E+00	0.00
15	1.00E-05	-	3.00E-05	3.90E+01	3.90E+01	3.90E+01	3.90E+01	3.90E+01	3.90E+01	3.90E+01	3.88E+01	3.87E+01	0.00
16	3.00E-05	-	1.00E-04	2.45E+02	2.45E+02	2.45E+02	2.45E+02	2.45E+02	2.46E+02	2.46E+02	2.44E+02	2.44E+02	0.00
17	1.00E-04	-	5.50E-04	3.49E+03	3.49E+03	3.49E+03	3.49E+03	3.49E+03	3.49E+03	3.49E+03	3.48E+03	3.47E+03	0.00
18	5.50E-04	-	3.00E-03	4.44E+04	4.44E+04	4.44E+04	4.44E+04	4.44E+04	4.44E+04	4.44E+04	4.42E+04	4.41E+04	0.00
19	3.00E-03	-	1.70E-02	5.99E+05	5.99E+05	5.99E+05	5.99E+05	5.99E+05	6.00E+05	6.00E+05	5.97E+05	5.95E+05	0.00
20	1.70E-02	-	1.00E-01	8.36E+06	8.36E+06	8.36E+06	8.36E+06	8.37E+06	8.37E+06	8.37E+06	8.34E+06	8.30E+06	0.01
21	1.00E-01	-	4.00E-01	5.68E+07	5.68E+07	5.68E+07	5.68E+07	5.69E+07	5.69E+07	5.69E+07	5.66E+07	5.64E+07	0.07
22	4.00E-01	-	9.00E-01	1.24E+08	1.24E+08	1.24E+08	1.24E+08	1.24E+08	1.24E+08	1.24E+08	1.24E+08	1.23E+08	0.16
23	9.00E-01	-	1.40E+00	1.24E+08	1.24E+08	1.24E+08	1.24E+08	1.24E+08	1.24E+08	1.24E+08	1.23E+08	1.23E+08	0.16
24	1.40E+00	-	1.85E+00	9.84E+07	9.84E+07	9.84E+07	9.84E+07	9.84E+07	9.85E+07	9.85E+07	9.81E+07	9.77E+07	0.13
25	1.85E+00	-	3.00E+00	1.81E+08	1.81E+08	1.81E+08	1.81E+08	1.81E+08	1.81E+08	1.81E+08	1.80E+08	1.79E+08	0.23
26	3.00E+00	-	6.43E+00	1.65E+08	1.65E+08	1.65E+08	1.65E+08	1.65E+08	1.65E+08	1.65E+08	1.64E+08	1.64E+08	0.21
27	6.43E+00	-	2.00E+01	1.57E+07	1.57E+07	1.57E+07	1.57E+07	1.57E+07	1.57E+07	1.57E+07	1.57E+07	1.56E+07	0.02
			total	7.73E+08	7.73E+08	7.73E+08	7.73E+08	7.73E+08	7.74E+08	7.74E+08	7.70E+08	7.68E+08	1.00

Table 5. Spontaneous Fission Neutron Spectrum from Spent Fuel

Fig. 4 indicates the neutron total cross section for Al^{27}, C^{12}, B^{11}, B^{10}, Li^{7}, He^{4} in irradiated neutron absorbing material, as shown in Fig. 4, between 0.1MeV and 6.4MeV, cross sections for the nuclides except C^{12} is almost same - 1b ~ 10 b. However, in case of B^{10}, at 0.0253 eV, total cross section is 3840 b and at 14 MeV, it is 1.467 b, and for Al^{27}, C^{12}, B^{11} , at 0.0253 eV, 1.65 b ~ 5.1 b and at 14 MeV, 1.3 b ~ 1.7 b. So, B^{10} content depends upon not the fast neutrons emitted from spent fuels close to the neutron absorbing material but the thermalized neutrons far away from it.

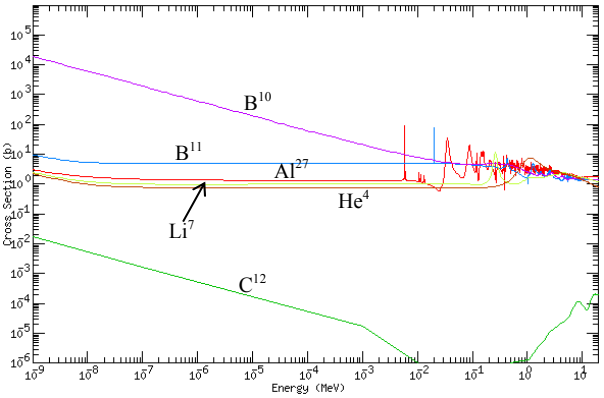

Fig. 4. Neutron Total Cross Section for Major Nuclides

Considering neutron mean free path in water (assuming 1 mole) - about 0.66 cm at 0.0253 eV and 9.98 cm at 14 MeV and general design of spent fuel storage facility shown in Fig.5, two spent fuels share one neutron absorbing material with distance of 4.7 mm to 10 mm between outermost fuel rod and neutron absorbing material, it is difficult for fast neutrons produced from spent fuels directly to react with neutron absorbing material close to them.

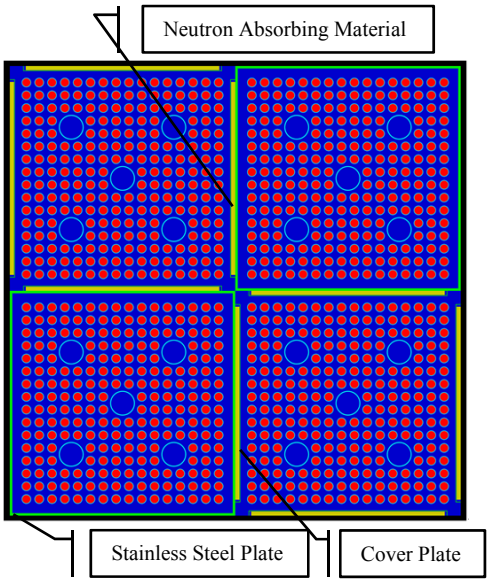

Fig. 5. General Design of Spent Fuel Storage Facility Unit Cell[4]

Fig. 6 shows the radiative capture cross section for major nuclides. Comparing to the total cross section on Fig. 4, radiative capture cross section for B^{10} is linearly decreased to 1 MeV of neutron energy and it is sharply declined below 10^{-6} b above 1 MeV. For other nuclides, it is linearly decreased except resonance region around 0.01 MeV to 1 MeV. Radiative capture cross section for major nuclides is between 10^{-6} b to 1 b. Therefore, for

thermal neutrons, radiative capture reaction could be happened for B^{10} and Al^{27}.

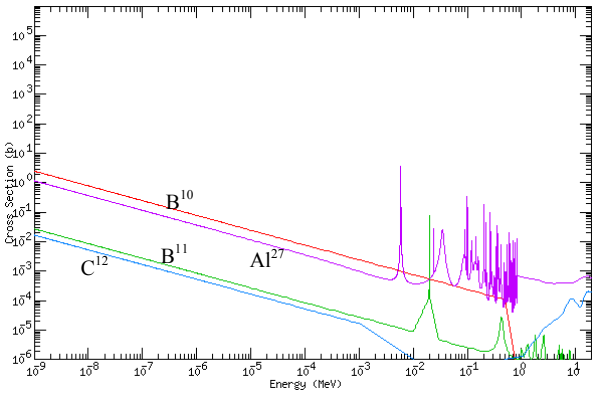

Fig. 6. Radiative Capture Cross Section for Major Nuclides

To confirm the variation of B^{10} content, dominant neutron absorption cross sections for B^{10} , radiative capture, (n, p), (n, d), (n, α) and (n, t2a), should be identified. As shown in Fig. 7, most dominant reaction between B^{10} and neutron is (n, α) for every energy range, especially, for thermal regions, cross section is around 10^4 b. Besides, around high energy range, from 0.1 MeV to 10 MeV, it is possible to happen other neutron absorption reactions with cross section of 10^{-6} b to 10^{-1} b.

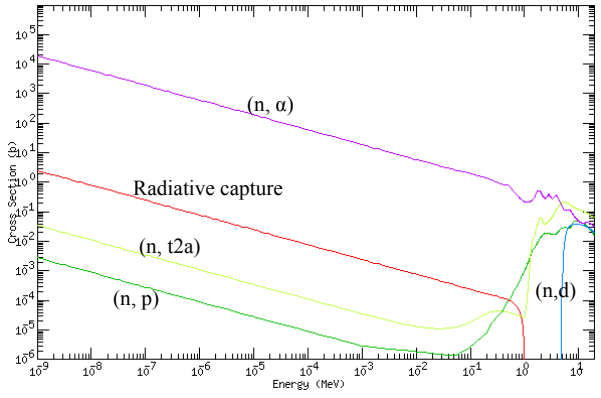

Fig. 7. Neutron Absorption Cross Section for B^{10}

If neutron absorbing material is irradiated by neutrons over 40 years continuously, B^{10} content could be exponentially decreased as per the formula (2.2.1), and it becomes to be more decreased as the irradiation period is longer and neutron flux is larger. Furthermore, it strongly depends on the neutron absorption cross section. So, the most dominant reaction to impact the B^{10} content is (n, α).

$$\frac{dN(t)}{dt} = -\sigma_c \phi N(t) \qquad (2.2.1)$$

$$N(t) = N_0 e^{-\sigma_c \phi t}$$

N_o : Initial B^{10} content
σ_c : cross section (barn)
ϕ : neutron flux (n/cm^2-sec)
t : irradiation time (minimum 40 years)

2.3 B^{10} Production

B^{10} could be produced by radioactive decay of parent nuclides of Be^{10} with half-life of 1,510,000 years as beta decay and C^{10} with half-life of 19.255 sec as electron capture. Besides, as shown in Fig. 8, B^{11} has 0.1 b for high energy neutron absorption cross section, (n, 2n), and it may be changed to B^{10}.

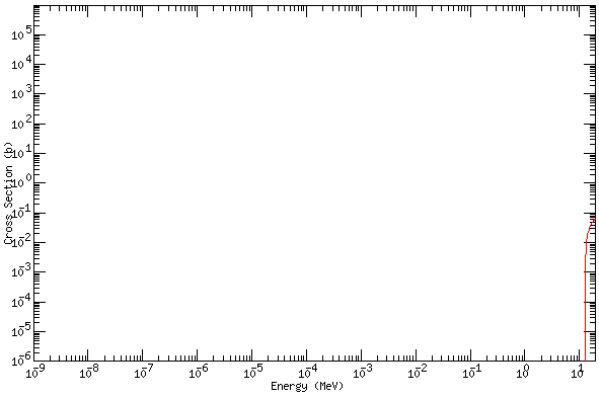

Fig. 8. (n, 2n) Cross Section for B^{11}

2.4 Minimum Neutron Flux for Variation of B^{10} Content

As described in Section 2.2, 10^8 order of neutron flux is not sufficient to change the nuclide composition in neutron absorbing material, especially B^{10} content and it is necessary to identity the minimum neutron flux changing the nuclide composition in material.

Table 6. Nuclide Composition Variation in Neutron Absorbing Material as per the Neutron Flux Intensity

Nuclide	Weight Percent (%)				
	Before Irradiation	Flux Irradiation(n/cm^2-s)			
		10^8	10^9	10^{10}	10^{11}
al27	72.01	72.02	72.01	71.99	71.84
b11	17.87	17.85	17.85	17.84	17.8
c12	6.08	6.01	6.01	6.00	5.99
b10	4.04	4.05	4.02	3.70	1.63
c13	-	0.07	0.07	0.07	0.07
li7	-	-	0.03	0.25	1.69
he4	-	-	0.01	0.15	0.98
Total	100.00	100.00	100.00	100.00	100
k_{eff}	0.88828	0.88704	0.88705	0.89011	0.92385

So, neutron flux is intentionally increased from 10^8 order to 10 ^{11} order. Table 6 describes the comparison of nuclide composition variation in material as per the increasing neutron flux. As the neutron flux is increased, B^{10} content is decreased and C^{13}, Li^7, He^4 contents are increased. Especially, B^{10} content is rapidly decreased from 10^{11} order of neutron flux. However, the content of Al^{27}, C^{12}, B^{11} is not decreased rapidly with the

increase of neutron flux. So, above 10^9 neutron flux, B^{10} begins to decrease and is changed to Li^7 and produce He^4 nuclides as described in formula (2.4.1).

$$B_5^{10} + n_0^1 \rightarrow Li_3^7 + \alpha\,(He_2^4) \qquad (2.4.1)$$

2.5 Neutron Multiplication Factor Calculation

Neutron multiplication factor is calculated to identify the neutron absorbing ability variation in neutron absorbing material caused by the neutron irradiation.

As shown in Fig. 5, spent fuel storage facility is modeled with four spent fuels of U^{235} 5.0 wt% - 42,700MWD/MTU with periodic boundary condition. The basic unit constituting a spent fuel storage facility is a structure called a cell, which is a square pillar of stainless steel plate that encloses one spent fuel with neutron absorbing material. Spent fuel storage facility is a welded honeycomb structure which consists of a square cross-section of cells. Each cell is designed to accommodate a single spent fuel and neutron absorbing material is only used for maintaining the criticality safety. On the outer surface of stainless steel plate, another thin thickness of stainless steel plate called cover-plate is attached to support the neutron absorbing material. So, each neutron absorbing material is inserted between two stainless steel plates. All neutron absorbing materials cover the full length of active fuel region. Except spent fuel, stainless steel plate and neutron absorbing material, it is assumed that all spaces are filled with water.

Nuclide composition of neutron absorbing material applies the result induced in Section 2.2. Table 6 describes the comparison of neutron multiplication factor between before and after irradiation conditions. Until neutron flux of 10^9 order, there is little difference on neutron multiplication factor caused by the little difference on B^{10} content. However, from neutron flux of 10^{10} order, neutron multiplication factor is increased caused by the decrease of B^{10} content.

3. Conclusions

The minimum neutron flux to impact the neutron absorbing ability is 10^{10} order, however, usual neutron flux from spent fuel is 10^8 order. Therefore, even though neutron absorbing material is irradiated for over 40 years, B^{10} content is little decreased, so, initial neutron absorbing ability could be kept continuously. NUREG-1536[5] and NUREG-1617[6] regulate obtaining 25% margin of B^{10} content in neutron absorbing material, and considering the neutron flux from spent fuel, it is enough to maintain the criticality safety in spent fuel storage facility.

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