A Study on Effect of Assembly Power History to Gamma Dose Rate in Spent Fuel Pool at a Short Cooling Time

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1. Introduction

Spent fuel assemblies having different burnups are stored in the spent fuel pool (SFP) during refueling period of nuclear reactor. The radiation dose rate on the water surface of SFP must be less than 25 $\mu\mathrm{Sv/hr}$ according to ANSI/ANS 57.1-1992 [1, 2]. The radiation source terms of spent fuel can be affected by many parameters such as fuel enrichment, assembly burnup, power history and cooling time. In this study, tendency of radiation source terms depending on burnup and power history is observed in 2-D fuel assembly and 3-D whole core at a short cooling time of 100 hours to consider a refueling situation. Dose rates are calculated for a single fuel assembly in transit inside a SFP. The effect of spatial gamma source distribution to the dose rate is also studied by using a fuel assembly having a flat gamma source distribution. The required water depth from the fuel assembly in transit inside a SFP is also evaluated to make the dose rate at a water surface less than 25 $\mu\mathrm{Sv/hr}$.

2. Methods and Results

2.1 Codes and Methods

A deterministic neutron transport code STREAM is used to calculate radiation source term. It has a capability to compute radioactivity, decay heat, neutron and gamma source spectra for spent nuclear fuel [3]. STREAM calculates gamma source terms corresponding to decay X-ray, decay gamma, spontaneous gamma, (α, n) reaction gamma and bremsstrahlung. It is also used to depletion calculation of fuel assemblies in Section 2.2.

A Monte Carlo particle transport code MCS is used to 3-D whole core depletion calculation and gamma dose rate calculation in a SFP. A fixed source mode simulation is used with a weight window (WW) variance reduction technique [4] for the gamma dose rate calculation. The WW method is a widely used variance reduction technique for a shielding calculation in Monte Carlo code.

if $w > w_{\mathrm{upper}}$ **then**
  Split the particle
else if $w < w_{\mathrm{lower}}$ **then**
  Play Russian roulette
end if

Fig. 1. Weight window method. w is a particle weight, w_{upper} and w_{lower} are upper and lower weight window boundaries, respectively.

Fig. 1 shows how particles are treated during the WW simulation. In the WW simulation, particles are split or removed depending on the WW boundaries which are usually defined by an importance function to a target response. As a result, more particles are simulated near the target tally position. Additionally, 10 statistical tests implemented in MCS are used to assess a validity of tally results [5].

The radiation source term results of 3-D hypothetical OPR-1000 core are presented in Section 2.3. Gamma dose rates in a SFP are presented in Section 2.4. The gamma dose rates are calculated for a single fuel assembly in transit which is one of fuel assemblies from the hypothetical core. Fig. 2 shows the calculation procedure for section 2.3 and section 2.4. A 3-D whole core depletion calculation is done prior to the source term calculation. A Monte Carlo neutron transport code MCS is used with ENDF/B-VII.1 nuclear data library. In the depletion calculation, 20 axial meshes are applied to every fuel rod. Additionally, depletion zones of gadolinia pins are radially divided into 10 rings to consider a spatial self-shielding effect of it.

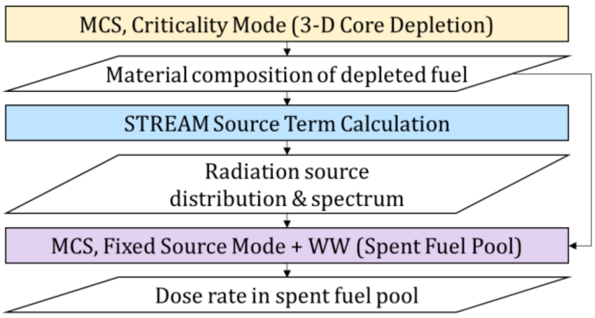

Fig. 2. Calculation procedure for dose rate calculation.

The material composition of depleted fuels at the end of cycle of hypothetical core is transferred to STREAM. Then, STREAM computes spatial source distribution and energy spectrum for neutron and gamma. The radiation source terms are transferred to MCS to define initial sources used in a SFP calculation. The material composition of depleted fuels from 3-D whole core depletion calculation is also used in the SFP calculation. Additional details are presented in Section 2.4.

2.2 Radiation Source Term Calculation for 2-D Fuel Assembly

Radiation source term calculation is done by STREAM for 2-D PLUS7 fuel assembly. Enrichments of 4.65 and 4.10 wt% are used for normal and zoned fuel rods, respectively. The tendency of neutron and gamma source intensities in a spent nuclear fuel is studied through two numerical tests: 1) Different assembly burnup with the same power density, and 2) Same assembly burnup with different power history.

In the first numerical test, radiation source term is calculated for assemblies having different burnup. The same power density of 36.85 W/g is used up to each target burnup. Fig. 3 shows the radiation source term results. It is observed that neutron source intensity is proportional to the assembly burnup. On the other hand, there are relatively small differences in gamma source intensity among assemblies having different burnup.

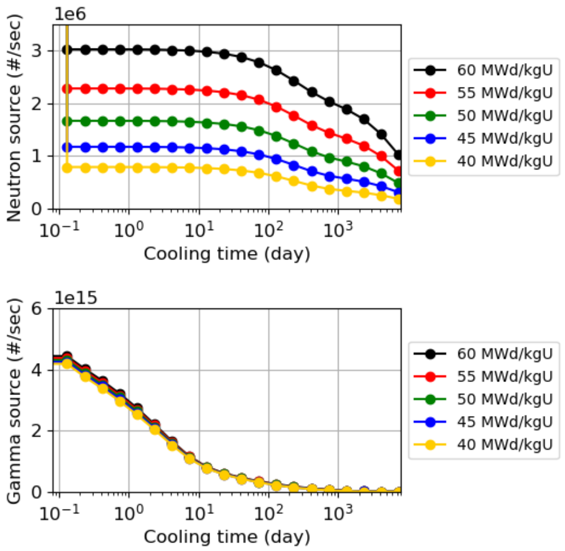

Fig. 3. Neutron and gamma source intensities of 2-D fuel assembly with different assembly burnup and same power density.

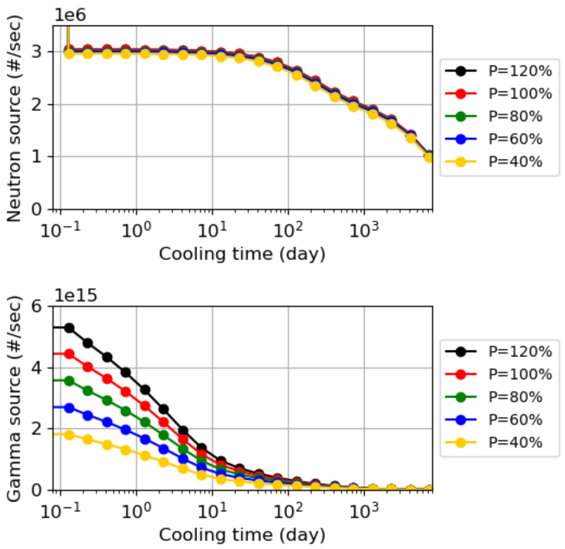

Fig. 4. Neutron and gamma source intensities of 2-D fuel assembly with different power history and same burnup.

In the second numerical test, it is assumed that each assembly is burned up to 40 MWd/kgU with the same power density of 36.85 W/g. After 40 MWd/kgU, the power density levels are changed to 40, 60, 80, 100%, and 120% of 36.85 W/g, respectively, and assemblies are burned up to 60 MWd/kgU. The different power density levels in 40–60 MWd/kgU represent the different assembly power of twice burned assemblies in the core. Fig. 4 shows the neutron and gamma source intensities with different power history. There are small differences in neutron source intensities depending on assembly power history, whereas gamma source intensities are highly proportional to it.

2.3 Radiation Source Term Calculation for 3-D Core Model

A 3-D whole core depletion calculation is done by MCS for a hypothetical OPR-1000 core to provide realistic material compositions of spent nuclear fuels. Fig. 5 shows assembly-wise burnup and normalized power distribution at the end of cycle of hypothetical core.

FA-wise Burnup @ EOC (MWd/kgU)

	H	J	K	L	M	N	P	R
1	49.1	46.5	47.7					
2	20.8	20.9	20.2	15.7	47.8			
3	43.3	40.7	23.5	52.5	17.2	49.4		
4	44.6	24.2	44.5	36.6	21.9	17.2	47.8	
5	23.9	44.0	38.9	42.6	36.6	52.5	15.7	
6	43.4	41.8	23.7	38.9	44.5	23.5	20.2	47.7
7	41.0	23.0	41.8	44.0	24.2	40.7	20.9	46.5
8	33.8	41.0	43.4	23.9	44.6	43.3	20.8	49.1

FA-wise Power @ EOC

	H	J	K	L	M	N	P	R
1	0.48	0.48	0.39					
2	1.21	1.21	1.14	0.90	0.39			
3	1.07	1.14	1.30	0.88	1.02	0.44		
4	1.09	1.33	1.05	1.11	1.25	1.02	0.39	
5	1.32	1.08	1.12	1.05	1.11	0.88	0.90	
6	1.04	1.10	1.32	1.11	1.05	1.30	1.14	0.39
7	1.07	1.30	1.10	1.08	1.33	1.14	1.21	0.48
8	0.85	1.07	1.04	1.32	1.09	1.07	1.21	0.48

Fig. 5. Assembly-wise burnup and normalized power distribution at end of cycle.

Fig. 6 shows the assembly-wise radiation source distribution calculated by STREAM. Cooling time of 100 hours is applied to the source term calculation. As it is observed in the previous section, it is clearly observed that neutron and gamma source distributions highly depend on burnup and power distributions, respectively. Fig. 7 shows the assembly-wise radiation source distribution at cooling time of 20 years. The shape of neutron source distribution is almost same with the one at cooling time of 100 hours, even though the magnitudes are decreased. On the other hand, the shape of gamma source distribution is changed a lot compared with the one at cooling time of 100 hours. At cooling time of 20 years, it is highly dependent on burnup distribution. At a short cooling period, the gamma source emission is largely affected by short-lived nuclides in fission products. On the other hand, after few years cooling period, most short-lived gamma emitting nuclides disappear, and the gamma emission is mostly affected by long-lived fission products such as Cs-137 ($T_{1/2}$ = 30.1 y).

Neutron source (1E+08/sec), Tcool = 100 hrs

	H	J	K	L	M	N	P	R
1	6.61	5.38	5.70					
2	0.36	0.37	0.33	0.15	5.84			
3	4.78	3.85	0.53	8.42	0.20	6.71		
4	5.25	0.60	5.19	2.50	0.43	0.20	5.84	
5	0.57	4.97	3.16	4.38	2.50	8.41	0.15	
6	4.87	4.15	0.57	3.16	5.19	0.53	0.33	5.70
7	3.90	0.50	4.15	4.97	0.60	3.85	0.37	5.38
8	5.27	3.90	4.87	0.57	5.25	4.78	0.36	6.61

Gamma source (1E+17/sec), Tcool = 100 hrs

	H	J	K	L	M	N	P	R
1	2.98	2.95	2.46					
2	6.38	6.37	5.99	4.60	2.44			
3	6.28	6.60	6.95	5.32	5.24	2.72		
4	6.47	7.15	6.23	6.27	6.56	5.24	2.44	
5	7.08	6.36	6.40	6.13	6.27	5.32	4.59	
6	6.12	6.38	7.04	6.40	6.23	6.95	5.98	2.46
7	6.17	6.92	6.38	6.36	7.15	6.60	6.38	2.95
8	5.16	6.17	6.12	7.08	6.47	6.28	6.38	2.98

Fig. 6. Assembly-wise radiation source distribution at cooling time of 100 hours.

Neutron source (1E+08/sec), Tcool = 20 years

	H	J	K	L	M	N	P	R
1	2.16	1.67	1.88					
2	0.06	0.06	0.05	0.02	1.93			
3	1.35	1.04	0.10	2.70	0.03	2.23		
4	1.53	0.11	1.51	0.60	0.07	0.03	1.93	
5	0.11	1.43	0.82	1.23	0.60	2.70	0.02	
6	1.38	1.14	0.11	0.82	1.51	0.10	0.05	1.88
7	1.04	0.09	1.14	1.43	0.11	1.04	0.06	1.67
8	1.60	1.04	1.38	0.11	1.53	1.35	0.06	2.16

Gamma source (1E+15/sec), Tcool = 20 years

	H	J	K	L	M	N	P	R
1	3.10	2.96	3.03					
2	1.45	1.46	1.41	1.11	3.02			
3	2.82	2.69	1.62	3.34	1.21	3.09		
4	2.89	1.66	2.88	2.45	1.52	1.21	3.02	
5	1.64	2.87	2.58	2.77	2.45	3.34	1.11	
6	2.81	2.75	1.63	2.58	2.88	1.62	1.41	3.03
7	2.70	1.59	2.75	2.87	1.66	2.69	1.46	2.96
8	2.10	2.70	2.81	1.64	2.89	2.82	1.45	3.10

Fig. 7. Assembly-wise radiation source distribution at cooling time of 20 years.

2.4 Dose Rate Calculation in Spent Fuel Pool

MCS fixed source mode simulation and weight window technique with 10 statistical tests [5] are used for the dose rate calculation in a SFP. Fig. 8 shows a SFP modeled with MCS and a WW boundary map. A SFP consists of a concrete structure of 1 meter thickness and is filled with water inside it. There is no soluble boron in the water of SFP. More than a thousand fuel assemblies can be stored at the bottom of SFP with a water shielding depth of ~8 meters over the active fuel of them. It is assumed that a single fuel assembly is in transit as shown in Fig. 8. One of three assemblies located at M07, K08 and P04 of hypothetical core shown in Fig 5 is selected as the assembly in transit. M07 and P04 assemblies represent the one having maximum and minimum gamma source intensities in the core, respectively. Cooling time of 100 hours is applied to the assembly in transit to consider a fuel assembly handling situation during a refueling period.

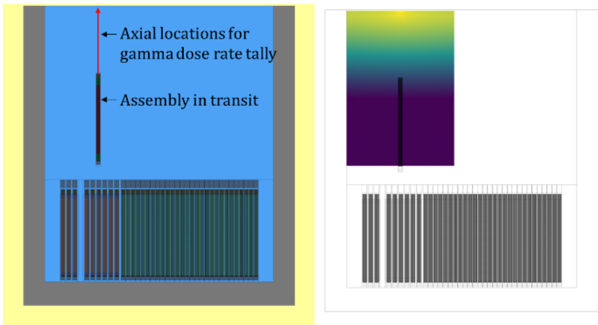

Fig. 8. Cross-sectional view of spent fuel pool model (left) and importance map applied to simulation (right).

The gamma source distribution and spectrum are calculated by STREAM and used in MCS to define initial photon source. Fig. 9 shows the axial gamma source distribution of three selected assemblies. The gamma source intensities are much lower at top/bottom ends of assembly due to lower axial power in axial cutback and blanket regions. K08* represents an assembly having the total gamma source intensity of K08 without axial source distribution. It is added to check the effect of axial gamma source distribution. Fig. 10 shows the gamma source spectrum of three selected assemblies at cooling time of 100 hours. The gamma source spectrum shapes of three selected assemblies are almost same with each other. It is observed that the gamma source emission at cooling time of 100 hours is mostly caused by short-lived nuclides. About 68% of gamma sources are emitted below 450 keV. Most of the low energy photons are caused by Np-239 ($T_{1/2}$ = 2.36 d), which is produced in a transmutation process from U-238 to Pu-239. The spectrum peak at 450–500 keV is caused by Ru-103 ($T_{1/2}$ = 39.2 d) and La-140 ($T_{1/2}$ = 1.68 d). The spectrum peak at 700–800 keV is caused by Nb-95 ($T_{1/2}$ = 35 d) and Zr-95 ($T_{1/2}$ = 64 d).

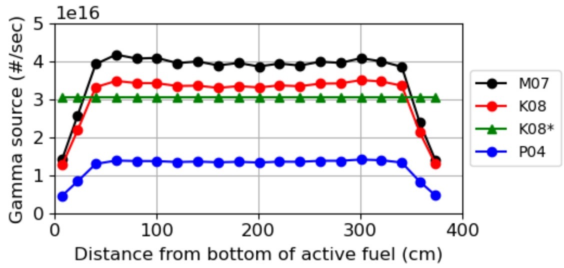

Fig. 9. Axial gamma source distribution of selected assemblies at cooling time of 100 hours.

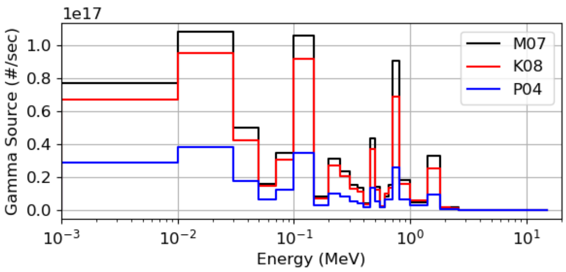

Fig. 10. Gamma source spectrum of selected assemblies at cooling time of 100 hours.

The gamma dose rate is calculated by multiplying photon flux and photon flux-to-dose conversion factor with log-log interpolation. The conversion factor from ICRP-116 [6] is used with log-log interpolation. Fig. 11 shows the conversion factor.

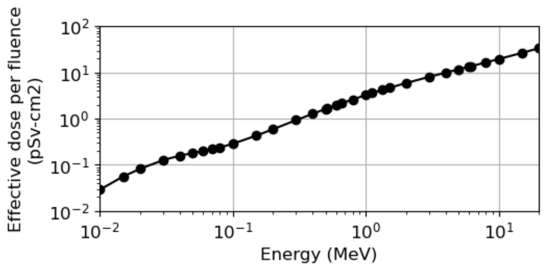

Fig. 11. Photon flux to dose rate conversion factor from ICRP-116 [6].

During the gamma dose rate calculations, it is found that a gamma dose rate is decreased to ~10^{-5} of it when there is a water shielding of 2 meters. Therefore, the gamma dose rate on the water surface from assemblies stored at bottom of SFP would be decreased to ~10^{-20} of it which is negligible enough due to a water shielding depth of ~8 meters. Therefore, it is decided to locate and sample the initial gamma source only in the assembly in transit. The dose rate by neutron and secondary gamma sources are neglected.

During the MCS photon simulation in SFP, dose rates are tallied at axial locations above the assembly in transit. Fig. 12 shows dose rate results for selected fuel assemblies in SFP. Yellow horizontal line represents the dose rate limit of 25 $\mu\mathrm{Sv/hr}$. Black vertical dashed line represents the required water depth, which is 284.38 cm, to make the dose rate on water surface less than 25 $\mu\mathrm{Sv/hr}$. It should be noted that all tally results passed 10 statistical tests.

The dose rate from selected fuel assemblies exponentially decreases as tally position is further from top of active fuel due to a water shielding. As mentioned previously, M07 and P04 assemblies have the maximum and minimum gamma source intensities in the hypothetical core, respectively. Therefore, if it is assumed that the shape of axial gamma source distribution is similar among assemblies in the core, the light purple colored region between M07 and P04 represents available gamma dose rate for the other assemblies. It is observed that dose rates from K08 are included in the region. The gamma dose rates from the other assemblies in the octant of hypothetical core are also plotted together as grey lines. It should be noted that the gamma dose rates from the other assemblies are also included in the light purple region between M07 and P04.

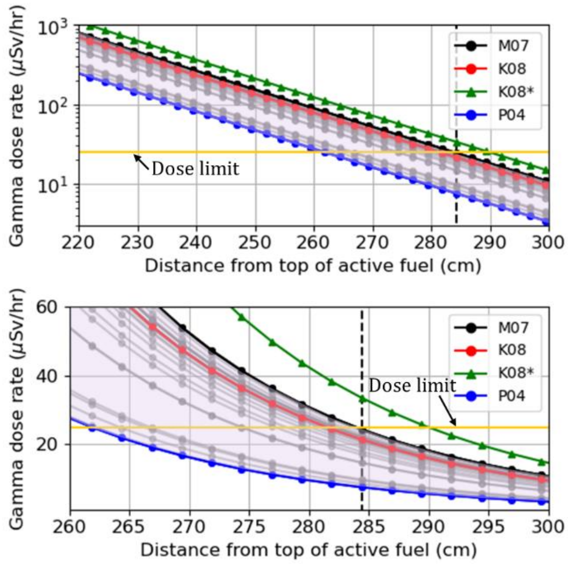

Fig. 12. Dose rate of selected fuel assemblies.

The dose rate from K08* assembly is also presented in Fig. 12. Even though the total gamma source intensities of K08 and K08* are same, the dose rate from K08* is ~1.6 time higher than the one from K08. It is much higher than the one from M07 which has ~1.25 times bigger gamma source intensity. It is because of more gamma source emission at the top end of fuel assembly due to the flat source distribution of K08* shown in Fig. 9.

3. Conclusions

At a short cooling period, it is observed that neutron and gamma source intensities of spent nuclear fuel are highly dependent on burnup and power history, respectively. Therefore, not only assembly burnup, but also assembly power history should be carefully considered for the source term and dose rate calculations at a short cooling time such as a refueling period. Additionally, after a long cooling period of several years, it is observed that both neutron and gamma source intensities are highly affected by burnup rather than power history due to long-lived fission products.

Material composition of spent nuclear fuels are obtained from a 3-D whole core depletion calculation with MCS. The radiation source term is calculated by STREAM and is used as an initial source in a SFP calculation with MCS. Gamma dose rate is calculated by MCS photon simulation in a fixed source mode with WW technique. Assemblies having maximum and minimum gamma source intensities in the hypothetical core are located at transit position of SFP for gamma dose rate evaluation. It is found that water depth of 284.38 cm is required to make the dose rate on water surface less than 25 $\mu\mathrm{Sv/hr}$. The effect of axial gamma source distribution is also observed by comparing dose rates from K08 and K08* assemblies. The realistic axial gamma source distribution is low at the top/bottom ends of fuel assembly due to low axial power at axial cutback and blanket regions. Even though the total gamma source intensity are same between K08 and K08*, the gamma dose rate from K08*, which has a flat source distribution, is ~1.6 times higher than the one with realistic source distribution.

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