AN ASSESSMENT OF THE RADIATION DOSE RATE DUE TO AN OCCURRENCE OF THE DEFECT ON THE SPENT NUCLEAR FUEL ROD

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This study examines how much the radiation dose rate around it varies if a crack occurs on the spent nuclear fuel rod. The spent nuclear fuel rod to be examined is that of Kori unit 3&4. The source terms are evaluated using the ORIGEN-ARP that is part of the version 5.1 of the SCALE package. The radiation dose rate is assessed using the TORT. To check if the structure of a fuel rod is appropriately modeled in the TORT calculation, the calculation results by the TORT are compared with those by the ANISN for the same case. From the code simulation, it is known that if a crack occurs on the spent nuclear fuel rod, the neutron dose rate varies depending on what material is the crack filled with, but the gamma dose rate varies irrespective of type of the material that the crack is filled with.

Keywords : Spent Nuclear Fuel Rod, Crack, Source Term, Radiation Dose, Neutron Dose, Gamma Dose

1. INTRODUCTION

Enormous radioactive materials generated from nuclear reactions are included in the spent nuclear fuels (SNFs). Since those radioactive materials may cause fatal damage to the human being and the surrounding environment, the SNFs should be kept intact until they are safely stored and disposed. In compliance with the domestic regulations, hence, the integrity of SNFs shall be inspected periodically with the help of various equipment including radiation detector. In order to shorten the inspection period, hence, it is necessary to develop the methodology to inspect a SNF quickly and reliably with the occurrence and the location of a defect on the SNF being identified as soon as possible. Based on the concept that the occurrence and the location of a crack on SNF can be identified if the change in radiation dose rate around it is detectable, hence, it is examined how much radiation dose rate varies around it when a crack on SNF occurs, as a preliminary study for developing the above methodology.

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2. CALCULATION METHODS AND TOOLS

The fuel rod of Kori unit 3&4 is chosen for an analysis of the variation in radiation dose rate around a crack on SNF. The data for source term evaluation, geometry modeling and radiation dose assessment are obtained from the design documents for Kori unit 3 and 4 such as FSAR. The main design parameter values of the fuel rod of Kori unit 3&4 are summarized in Table 1. The structure of it is depicted in Fig. 1.

For source terms evaluation, the ORIGEN-ARP (Oak Ridge Isotope Generation and Depletion Code-Automatic Rapid Processing) [1], which is part of the version 5.1 of the SCALE (Standardized Computer Analyses for Licensing Evaluation) package [2], is used. For radiation dose assessment, the TORT (Three-Dimensional Discrete Ordinate Neutron/Photon Transport Code) [3] is used. For the reaction cross-section library, the BUGLE-96 [4] is used. The GIP program [5] is used to generate the input library file to the TORT. The dose conversion factors in ICRP-74 [6] are applied.

In this study, two different source terms are considered; neutron and gamma source terms. Neutron source terms consist of neutrons from (α, n) reactions of actinide nuclides

Table 1. The Design Parameter Values of the Fuel rod of Kori Unit 3 & 4.

Parameter	Value (cm)
fuel pellet radius	4.095
fuel gap thickness	0.085
fuel clad thickness	0.57
fuel rod total length	386.59
fuel rod (UO^2 region)	365.76
fuel rod (Plenum region)	18.517

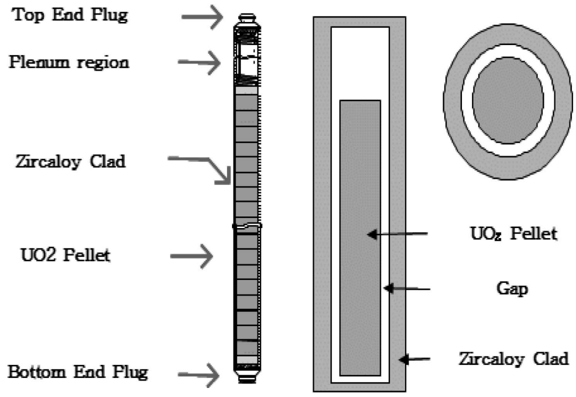

Fig. 1. The structure of a fuel rod.

Fig. 2. Neutron intensity per unit energy against neutron energy in UO_2 pellet region.

in the region of UO_2 pellet, spontaneous fission neutrons and delayed neutrons from radioactive decays of fission products. Gamma source terms consist of gamma rays from fission products in the region of UO_2 pellet and gamma rays from the activated zircaloy cladding. These source terms are evaluated and presented from Fig. 2 to Fig. 4.

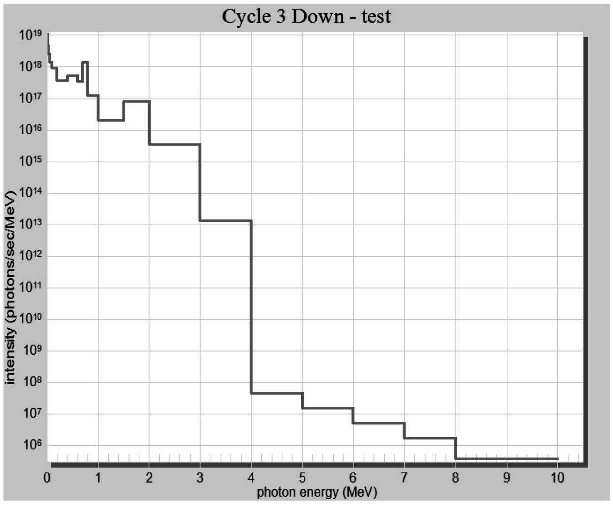

Fig. 3. Gamma intensity per unit energy against the gamma energy in UO_2 pellet region.

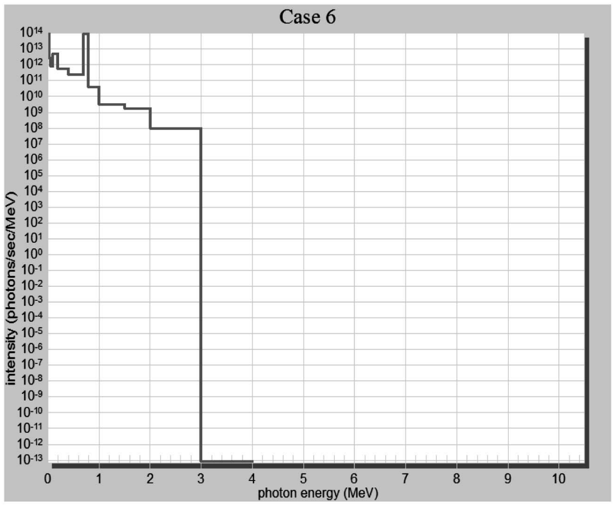

Fig. 4. Gamma intensity per unit energy against the gamma energy in the region of Zircaloy Clad.

Source term densities in unit of number/cm^3·sec that are inputs to the ANISN and the TORT are summarized in Table 2. These values are obtained by dividing source term strength (number/sec) by source volume (cm^3).

For the evaluation, the geometry of a fuel rod is modeled to consist of 3 regions: UO_2 pellet region; gap region; and zircaloy clad region, as shown in Fig. 5. The fuel rod is assumed to be surrounded by coolant. Considering symmetry of its geometry, only a quarter region of a fuel rod is modeled, as shown in Fig. 5. Regarding a state of the fuel rod, 3 cases are considered: Case (1) - normal without any crack; Case (2) - occurrence of a crack, which is filled with helium (He) gas that is the filling gas of the fuel gap; Case (3) - occurrence of a crack, which is filled with coolant. A crack is assumed to occur at the center of the

Table 2. Source Term Density (number /cm^3·sec).

Neutron Source Term (UO^2 Pellet Region)			
Neutron Group	Upper	Lower	[neutrons/cm^3·sec]
1	1.7732E+01	1.4191E+01	3.852E-03
2	1.4191E+01	1.2214E+01	1.723E-02
3	1.2214E+01	1.0000E+01	1.132E-01
4	1.0000E+01	8.6071E+00	2.824E-01
5	8.6071E+00	7.4082E+00	6.611E-01
6	7.4082E+00	6.0653E+00	1.945E+00
7	6.0653E+00	4.9659E+00	3.803E+00
8	4.9659E+00	3.6788E+00	1.101E+01
9	3.6788E+00	3.0119E+00	1.080E+01
10	3.0119E+00	2.7253E+00	6.014E+00
11	2.7253E+00	2.4660E+00	6.148E+00
12	2.4660E+00	2.3653E+00	2.574E+00
13	2.3653E+00	2.3457E+00	5.118E-01
14	2.3457E+00	2.2313E+00	3.074E+00
15	2.2313E+00	1.9205E+00	9.010E+00
16	1.9205E+00	1.6530E+00	8.517E+00
17	1.6530E+00	1.3534E+00	1.031E+01
18	1.3534E+00	1.0026E+00	1.290E+01
19	1.0026E+00	8.2085E-01	6.881E+00
20	8.2085E-01	7.4274E-01	2.952E+00
21	7.4274E-01	6.0810E-01	5.016E+00
22	6.0810E-01	4.9787E-01	3.980E+00
23	4.9787E-01	3.6883E-01	4.405E+00
24	3.6883E-01	2.9721E-01	2.265E+00
25	2.9721E-01	1.8316E-01	3.214E+00
26	1.8316E-01	1.1109E-01	1.670E+00
27	1.1109E-01	6.7379E-02	8.117E-01
28	6.7379E-02	4.0868E-02	3.903E-01
29	4.0868E-02	3.1828E-02	1.104E-01
30	3.1828E-02	2.6058E-02	6.308E-02
31	2.6058E-02	2.4176E-02	1.924E-02
32	2.4176E-02	2.1875E-02	2.249E-02
33	2.1875E-02	1.5034E-02	5.989E-02
34	1.5034E-02	7.1017E-03	5.366E-02
35	7.1017E-03	3.3546E-03	1.747E-02
36	3.3546E-03	1.5846E-03	5.669E-03
37	1.5846E-03	4.5400E-04	2.302E-03
38	4.5400E-04	2.1445E-04	2.837E-04
39	2.1445E-04	1.0130E-04	9.157E-05
40	1.0130E-04	3.7266E-05	3.440E-05
41	3.7266E-05	1.0677E-05	8.299E-06
42	1.0677E-05	5.0435E-06	1.008E-06
43	5.0435E-06	1.8554E-06	3.787E-07
44	1.8554E-06	8.7643E-07	7.437E-08
45	8.7643E-07	4.1399E-07	2.462E-08
46	4.1399E-07	1.0000E-07	1.070E-08
47	1.0000E-07	1.0000E-11	2.000E-09
Total			1.196E+02

Gamma Source Term (UO^2 Pellet Region)			
Gamma Group	Upper	Lower	[photons/cm^3.sec]
1	14	10	3.209E-03
2	10	8	6.772E-02
3	8	7	1.548E-01
4	7	6	4.602E-01
5	6	5	1.367E+00
6	5	4	4.058E+00
7	4	3	1.173E+06
8	3	2	3.148E+08
9	2	1.5	3.782E+09
10	1.5	1	8.984E+08
11	1	0.8	2.273E+09
12	0.8	0.7	1.246E+10
13	0.7	0.6	3.213E+09
14	0.6	0.4	9.401E+09
15	0.4	0.2	6.592E+09
16	0.2	0.1	8.521E+09
17	0.1	0.06	5.034E+09
18	0.06	0.03	6.847E+09
19	0.03	0.02	4.117E+09
20	0.02	0.01	9.880E+09
Total			7.333E+10

Gamma Source Term (Zircaloy Clad Region)			
Gamma Group	Upper	Lower	[photons/cm^3.sec]
1	14	10	0.000E+00
2	10	8	0.000E+00
3	8	7	0.000E+00
4	7	6	0.000E+00
5	6	5	0.000E+00
6	5	4	0.000E+00
7	4	3	5.525E-18
8	3	2	6.569E+03
9	2	1.5	5.580E+04
10	1.5	1	1.098E+05
11	1	0.8	5.361E+05
12	0.8	0.7	6.481E+08
13	0.7	0.6	1.732E+06
14	0.6	0.4	3.101E+06
15	0.4	0.2	7.315E+06
16	0.2	0.1	3.406E+07
17	0.1	0.06	2.096E+06
18	0.06	0.03	5.668E+06
19	0.03	0.02	6.313E+07
20	0.02	0.01	1.238E+07
Total			7.786E+08

fuel rod, whose size is 2 cm in length, 0.64 cm in width, and 0.57 cm in depth, as shown in Fig. 6.

To check if the structure of a fuel rod is appropriately modeled in the TORT code calculation, the calculation results by the TORT are compared with those by ANISN (One Dimensional Discrete Ordinate Transport Code with Anisotropic Scattering) [7,8] code for the Case (1). For this, first, 3 different reference points (r = 5, 10, 15 cm) distant

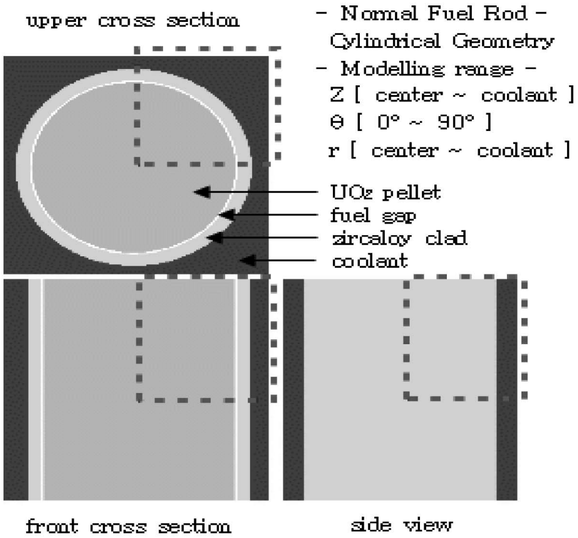

Fig. 5. The geometrical model of a fuel rod for the calculation by TORT.

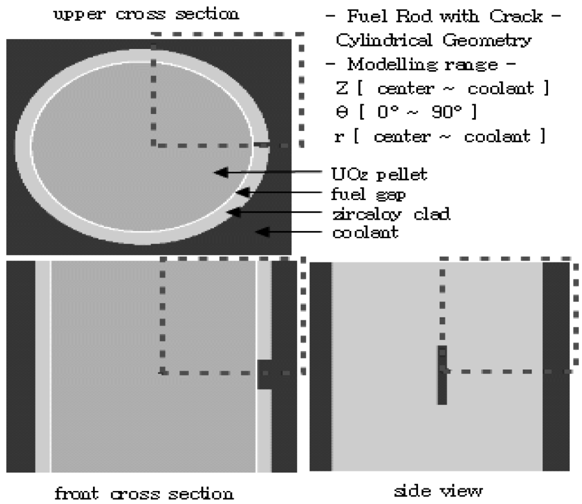

Fig. 6. The geometrical model of a crack on the fuel rod for the calculation by TORT.

Table 3. Summary of the Neutron Dose Rate Calculated at 3 Reference Points (r=5, 10, 15 cm). (unit: Sv/hr)

	(normal fuel rod)				
r	ANISN Result (Interpolation)	r_1	ANISN Calculation 1	r_2	ANISN Calculation 2
5	3.633E-04	4.94	3.7273E-04	5.32	3.1191E-04
10	7.017E-05	9.90	7.2078E-05	10.28	6.4999E-05
15	1.989E-05	14.85	2.0586E-05	15.23	1.8822E-05
	(normal fuel rod)				
r	TORT Result (Interpolation)	r_1	TORT Calculation 1	r_2	TORT Calculation 2
5	3.671E-04	4.69	4.212E-04	5.13	3.440E-04
10	7.050E-05	9.71	7.598E-05	10.47	6.176E-05
15	1.986E-05	14.28	2.354E-05	15.04	1.964E-05
	(crack filled with Helium)				
r	TORT Result (Interpolation)	r_1	TORT Calculation 1	r_2	TORT Calculation 2
5	3.806E-04	4.69	4.377E-04	5.13	3.562E-04
10	7.072E-05	9.71	7.622E-05	10.47	6.194E-05
15	1.989E-05	14.28	2.358E-05	15.04	1.966E-05
	(crack filled with Coolant)				
r	TORT Result (Interpolation)	r_1	TORT Calculation 1	r_2	TORT Calculation 2
5	3.537E-04	4.69	4.013E-04	5.13	3.333E-04
10	7.039E-05	9.71	7.584E-05	10.47	6.168E-05
15	1.985E-05	14.28	2.353E-05	15.04	1.962E-05

in radial direction from the center of a SNF are set. Second, 2 different points near each reference point are set, since there is a little difference in setting the calculation point in the TORT and the ANISN. Third, the neutron and the gamma dose rate at 2 different points near each reference point are calculated using the TORT and the ANISN, respectively. Then a total of 12 dose rate values, that is, 6 neutron dose rate values and 6 gamma dose rate values, are obtained by each code. Fourth, the neutron and the gamma dose rate at each reference point are obtained by interpolating the 2 dose rate values calculated at 2 different points near each reference point.

After this check, the neutron and the gamma dose rate at each reference point are obtained by interpolating the 2 dose rate values calculated using TORT at 2 different points near each reference point for Case (2) and Case (3).

3. RESULTS

The neutron and the gamma dose rate values calculated for Case (1)-(3) are summarized in Table 3 and 4, respectively. Comparison between the dose rate values in the first and the second row of Table 3 and 4 shows that they are almost in concord. It means that there is no major error in modeling a fuel rod for TORT code calculation.

Scrutiny of other calculation results in Table 3 and 4 shows that, the neutron and the gamma dose rate decrease as the calculation point is more distant in the radial direction from the SNF. Also, it is found that, at each reference point, the neutron dose rate for Case (2) is a little bigger, but the neutron dose rate for Case (3) is a little smaller than that for Case (1). The gamma dose rate at each reference point is always a little bigger than that for Case (1), irrespective of type of the material that the crack is filled with.

Regarding the behavior of variation in the neutron dose rate, as shown in Fig. 7 and Fig. 8, the neutron dose rate increases or decreases abruptly near the crack, depending on what material the crack is filled with. For Case (2) where the crack is filled with Helium gas, the neutron dose rate near the crack increases. However, for Case (3) where the crack is filled with coolant, the neutron dose rate near it decreases. Regarding the behavior of variation in the gamma dose rate, as shown in Fig. 9 and Fig. 10, the gamma dose

Table 4. Summary of the Gamma Dose rate Calculated at 3 Reference Points (r=5, 10, 15 cm). (unit: Sv/hr)

	(normal fuel rod)				
r	ANISN Result (Interpolation)	r_1	ANISN Calculation 1	r_2	ANISN Calculation 2
5	1.910E+02	4.94	1.942E+02	5.32	1.732E+02
10	7.446E+01	9.90	7.547E+01	10.28	7.171E+01
15	4.166E+01	14.85	4.230E+01	15.23	4.068E+01
	(normal fuel rod)				
r	TORT Result (Interpolation)	r_1	TORT Calculation 1	r_2	TORT Calculation 2
5	1.914E+02	4.69	2.136E+02	5.13	1.819E+02
10	7.475E+01	9.71	7.766E+01	10.47	7.011E+01
15	4.173E+01	14.28	4.497E+01	15.04	4.153E+01
	(crack filled with Helium)				
r	TORT Result (Interpolation)	r_1	TORT Calculation 1	r_2	TORT Calculation 2
5	2.148E+02	4.69	2.468E+02	5.13	2.012E+02
10	7.519E+01	9.71	7.817E+01	10.47	7.045E+01
15	4.179E+01	14.28	4.505E+01	15.04	4.159E+01
	(crack filled with Coolant)				
r	TORT Result (Interpolation)	r_1	TORT Calculation 1	r_2	TORT Calculation 2
5	2.106E+02	4.69	2.411E+02	5.13	1.975E+02
10	7.512E+01	9.71	7.808E+01	10.47	7.039E+01
15	4.178E+01	14.28	4.504E+01	15.04	4.158E+01

* Gamma dose in total = Gamma dose due to fission products in UO_2 pellet region + Gamma dose due to activation of zircaloy clad

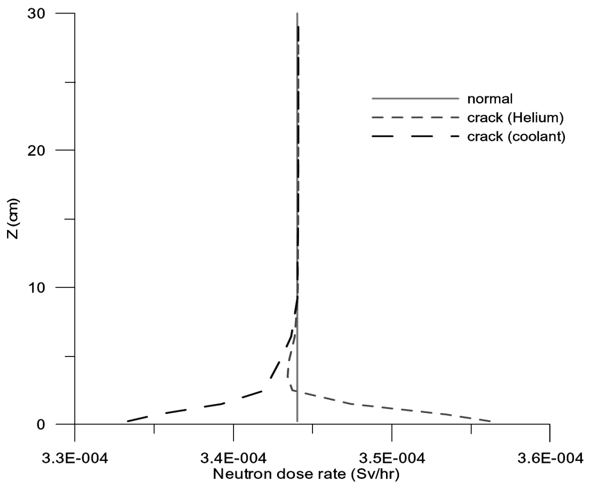

Fig. 7. The axial distribution of the neutron dose rate at r=5.13 cm distant from location of the crack. The solid line named as 'normal' is for Case (1) where the SNF rod is intact. The dotted line named as 'crack (Helium)' is for Case (2) where the crack is filled with Helium gas. The dashed line as 'crack (Coolant)' is for Case (3) where the crack is filled with coolant.

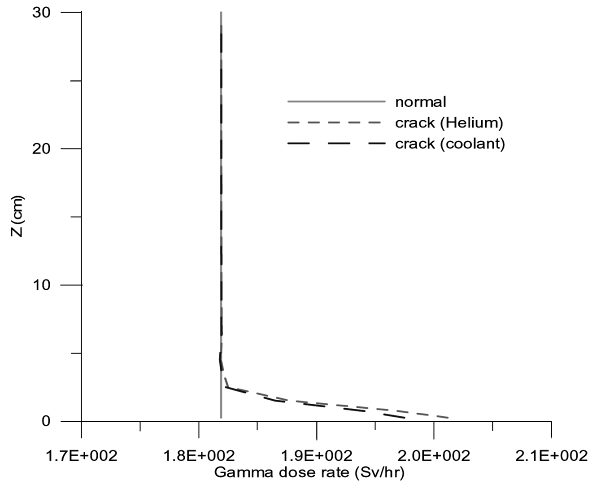

Fig. 9. The axial distribution of the gamma dose rate at r=5.13 cm distant from location of the crack.

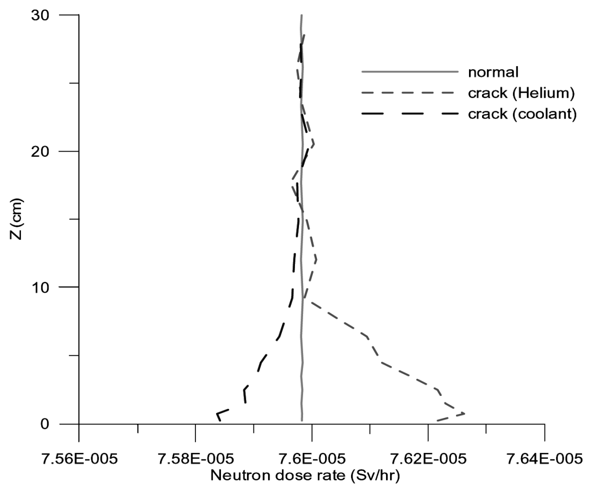

Fig. 8. The axial distribution of the neutron dose rate at r=9.7 cm distant from location of the crack.

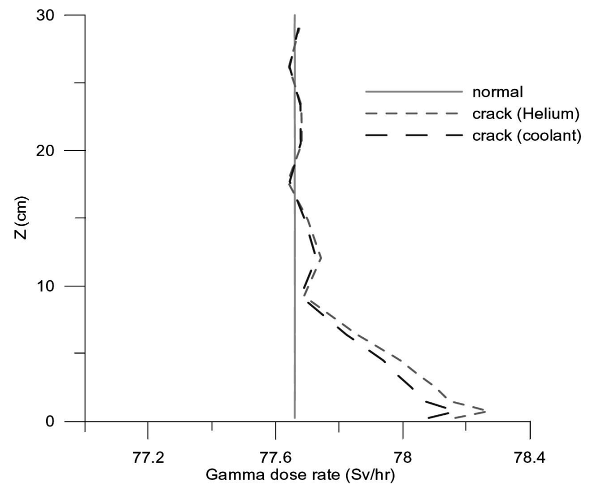

Fig. 10. The axial distribution of the gamma dose rate at r=9.7 cm distant from location of the crack.

Table 5. The Relative Variation in Radiation Dose Rate Due to a Crack on the SNF Rod.

The rate of variation in Neutron Dose			
r (cm)	Distance from the Surface of Fuel Rod ($r-r_1^*$) (cm)	crack (Helium)	crack (coolant)
5	0.25	3.60%	-3.70%
10	5.25	0.31%	-0.15%
15	10.25	0.15%	-0.05%
The rate of variation in Gamma Dose			
r	Distance from the Surface of Fuel Rod ($r-r_1$) (cm)	crack (Helium)	crack (coolant)
5	0.25	12.20%	10%
10	5.25	0.59%	0.49%
15	10.25	0.14%	0.11%

* r_1 is the outer diameter of the fuel rod of Kori unit 3&4, that is, 4.75 cm.

rate increases abruptly near the crack for both Case (2) and (3). Comparing between Case (2) and Case (3), the gamma dose rate in Case (2) is a little bigger than that in Case (3).

4. CONCLUSION

In this study, it is examined how much the radiation dose rate around it varies if a crack occurs on a SNF rod. By the computer code simulation, it is known that if a crack occurs on the SNF rod, the neutron dose rate near the crack varies differently depending on what material is the crack filled with. As shown in Fig. 7 and Fig.8, for Case (2) where the crack is filled with Helium gas, the neutron dose rate near the crack is more than that in Case (1), but for Case (3) where the crack is filled with coolant, the neutron dose rate near the crack is less than that in Case (1). As shown in Fig. 9 and Fig. 10, the gamma dose rate near the crack increases abruptly for both Case (2) and (3). Due to the occurrence of a crack, the relative variations in the radiation dose are summarized in Table 5.

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