RADIOLOGICAL DOSE ASSESSMENT ACCORDING TO METHODOLOGIES FOR THE EVALUATION OF ACCIDENTAL SOURCE TERMS

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The object of this paper is to evaluate the fission product inventories and radiological doses in a non-LOCA event, based on the U.S. NRC's regulatory methodologies recommended by the TID-14844 and the RG 1.195. For choosing a non-LOCA event, one fuel assembly was assumed to be melted by a channel blockage accident. The Hanul nuclear power reactor unit 6 and the CE 16×16 fuel assembly were selected as the computational models. The burnup cross section library for depletion calculations was produced using the TRITON module in the SCALE6.1 computer code system. Based on the recently licensed values for fuel enrichment and burnup, the source term calculation was performed using the ORIGEN-ARP module. The fission product inventories released into the environment were obtained with the assumptions of the TID-14844 and the RG 1.195. With two kinds of source terms, the radiological doses of public in normal environment reflecting realistic circumstances were evaluated by applying the average condition of meteorology, inhalation rate, and shielding factor. The statistical analysis was first carried out using consecutive three year-meteorological data measured at the Hanul site. The annual-averaged atmospheric dispersion factors were evaluated at the shortest representative distance of 1,000 m, where the residents are actually able to live from the reactor core, according to the methodology recommended by the RG 1.111. The Korean characteristic-inhalation rate and shielding factor of a building were considered for a series of dose calculations.

Keywords : Non-LOCA, Source term, Effective dose, TID-14844, RG 1.195

1. INTRODUCTION

The U.S. Nuclear Regulatory Commission (NRC) and its predecessor, the U.S. Atomic Energy Commission (AEC), have analyzed the source terms released into the environment from a maximum credible accident (MCA) and issued various versions of the regulatory guidelines including the technical information document (TID) – 14844 [1] and the regulatory guide (RG) 1.195 [2], which give acceptable assumptions in evaluating the radiological consequences. Based on these regulatory guides, the fission product inventory in a core, and possible release into the containment, has been evaluated for a construction permit or operating license of a nuclear reactor. However, each regulatory guide, with a different point of view, provides various acceptable methods and assumptions for the source term evaluations in several hypothetical design basis accidents (DBAs).

In particular, the TID-14844 issued in 1962, which has been widely used for evaluating the radiological consequences, suggests the assumptions of specific nuclides and their fraction released into the containment in a loss of coolant accident (LOCA). The RG 1.195, released in 2003, is basically in accordance with the regulation of the TID-14844 in the case of a LOCA event, and provides separate assumptions for determining the fission product inventory by a non-LOCA. Most of the regulatory authorities have recommended that the licensees performing a radiological assessment of the facility have complied with the methodology of the TID-14844 in a LOCA event and that of the RG 1.195 in a non-LOCA. For more conservative results, however, the radiological analysis for the non-LOCA is occasionally required to be assessed following the assumptions of TID-14844. In this case, the assumptions can lead to excessively severe conclusions than

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Table 1. Release Characteristics of Radionuclides Considered in TID-14844 and R.G. 1.195.

	TID-14844		**RG 1.195**	
Release point	Reactor building		Fuel gap	
Release traction	Noble gas	1.0	^{85}Kr	0.10
			Other noble gases	0.05
	Halogens	0.5 (with 0.5 plate-out)	^{131}I	0.08
			Other I	0.05
Decontamination factor			200 (noble gas = 1)	

expected, as the TID-14844 does not propose an additional methodology for evaluating a non-LOCA event [1-3]. To verify the difference and effects of the results assessed by two regulatory guides, the fission product inventory by a channel blockage accident, which is one of non-LOCA events, was evaluated in this study. Using two assessed source terms released into the environment, the effective doses of public, who is assumed to be influenced of the accident radiological doses in normal environmental/meteorological condition, were analyzed. The dose assessment was performed at the shortest representative distance of 1,000 m, where the residents are actually able to live from the nuclear reactor facility, with the average weather condition.

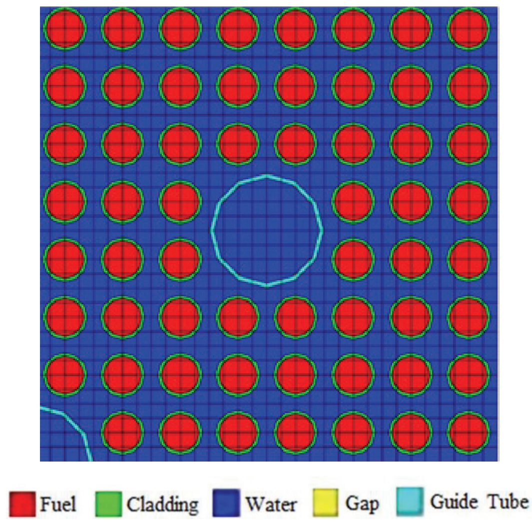

Fig. 1. A 1/4 fuel assembly model generated using the TRITON module.

2. MATERIALS AND METHODS

2.1 Accident Source Terms

The accident source term is considered to be an integral part of the design basis, since it sets forth a specific value or a range of values for controlling the parameters that constitute reference bounds for the design [4]. A number of regulatory guides suggest that the fission product inventory be determined using an appropriate isotope generation and depletion code such as the ORIGEN2 [5] or the ORIGEN-ARP [6]. Therefore, the procedure of the source term calculation in this study was divided into two parts: (1) the depletion calculation of the fuel assembly using the ORIGEN-ARP module of the SCALE6.1 computer code system, and (2) the analysis on the fission product inventory released into the containment based on the TID-14844 and the RG 1.195.

2.1.1 Depletion Calculations

To evaluate the source term released during a non-LOCA event, one fuel assembly was assumed to be melted by a channel blockage accident. The fission product inventory was analyzed based on the Hanul nuclear power reactor unit 6. The fuel assembly of Hanul unit 6 consists of a 16×16 array of 236 fuel rods and 5 guide tubes, and is closed at the top and bottom by end fittings [7]. The fuel has an enrichment of 4.51 wt% for 184 rods and 4.00 wt% for 52 rods, and an average burnup of 37,154 MWD/MTU.

The burnup-dependent cross section library for this fuel assembly model was generated using the TRITON module of the SCALE6.1 [8] (see Figure 1). The depletion calculations were performed using the NEWT module, which is a two-dimensional lattice physics code with discrete ordinates transport methods. The produced library was incorporated into the ORIGEN-ARP module to model the fuel assembly, and the inventory of the fission products was calculated based on a case of channel blockage accident.

2.1.2 Recommendation and Assumptions

The main release characteristics of nuclides recommended in the TID-14844 and the RG 1.195 are tabulated in Table 1. Two regulatory guides consider noble gases of krypton and xenon, and halogens of iodine as major fission products to be assessed. The TID-14844 assumes that 100% of the core inventory of noble gases and 50% of the iodines are released from the core into the containment, and half of the iodine in the containment is deposited onto the interior surfaces very rapidly. The RG 1.195, which applies a separate regulatory methodology on non-LOCA events, assumes that the fractions of the core inventory are in the gap. This shows that 10% of ^{85}Kr, 5% of other noble gases, 8% of ^{131}I, and 5% of other iodines will be released into the fuel gap and not into the containment vessel during a non-LOCA event. In addition, 99.5% of the total iodine released from the damaged rods is assumed to be retained by the water as this guide considers the decontamination effect. The retention of noble gases in the water is negligible.

Using the core inventory produced in section 2.1, the source terms released into the environment were calculated by reflecting the volume of the containment building and the leakage rate. Any effect by the additional filter system of the building was not considered in this study.

2.2 Dose Calculations

To compare the environmental impact assessment of the fission product release evaluated based on the TID-14844 and the RG 1.195, the effective doses of public were analysed at an arbitrary distance from the nuclear reactor. The public was assumed to be adults who are exposed on average by the accident radiation source in normal environmental condition. The effective dose was calculated as the sum of doses through the external exposure by a plume and the internal exposure by inhalation. The dose coefficients based on the International Commission on Radiological Protection (ICRP) Publication 60 for the nuclides were applied [9]. The external exposure dose by plume, $D_{E,pl}$, can be written as follows:

$$D_{E,pl} = \left(\frac{\chi}{Q}\right) SF \sum_i \frac{DC_{pl,i} \int_0^t Q_i dt}{3600}, \quad (1)$$

where (χ/Q) is the annual-averaged atmospheric dispersion factors (sec·m^{-3}),
SF is the shielding factor of the building,
$DC_{pl,i}$ is the dose coefficient for a plume on the i-nuclide (mSv-m^3/Bq-hr),
Q_i is the radioactivity of the i-nuclide released into the environment (Bq), and
3600 is the unit conversion constant (sec·hr^{-1}).

The effective dose by internal exposure, $D_{E,inh}$, was formulated by considering the chemical form of iodine, as denoted in Equation (2).

$$D_{E,inh} = \left(\frac{\chi}{Q}\right) R_{I,n} B \sum_i D_{inh,ia} \int_0^t Q_i dt, \quad (2)$$

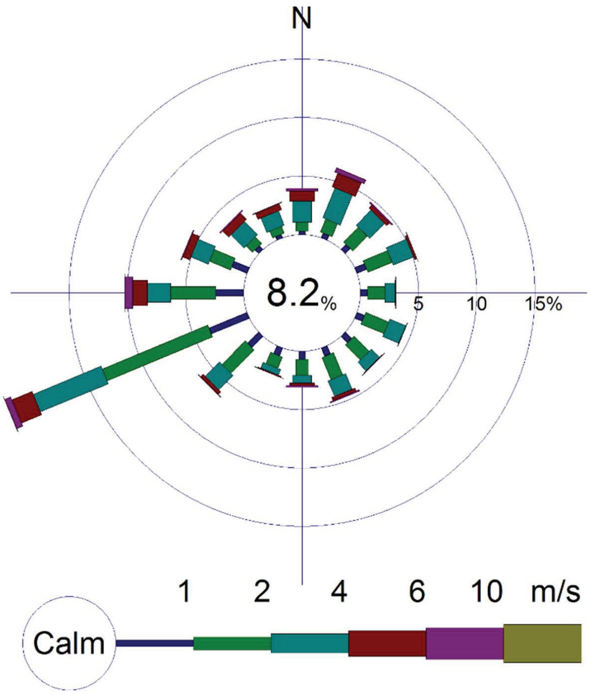

Fig. 2. Wind rose for the Hanul site obtained from 2006 to 2008.

where $R_{I,n}$ is the ratio of the chemical form of iodine, B is the annual-averaged inhalation rate (m^3·sec^{-1}), and $DC_{pl,i}$ is the dose coefficient for inhalation on the i-nuclide (mSv·Bq^{-1}).

A statistical analysis was carried out using consecutive three-year meteorological data measured at a 10 meter-height in the Hanul site from 2006 to 2008 (see Figure 2). To analyze the averagely-exposed radiological doses, the meteorological data in normal circumstances was used with the annual-averaged atmospheric dispersion factors, not the extreme condition such as the atmospheric dispersion factor of 99.5% and 95%. The atmospheric dispersion factor at the shortest representative distance of 1,000 m where the residents are actually able to live from the nuclear reactor was calculated through a classification by sixteen directions using the U.S. NRC's XOQDOQ computer code [10], which implements the methodology analysis with the assumptions recommended by the RG 1.111 [11]. Among the atmospheric dispersion factors of the sixteen directions, the maximum one, i.e., 2.82 × 10^{-5} sec·m^3 for the direction of ENE, was chosen as a representative value.

The inhalation rate was also applied under consideration of the domestic characteristics and normal condition of environment, and therefore the annual average value of 2.47 × 10^{-4} m^3·sec recommended by the domestic off-site dose assessment [12]. In addition, the

Table 2. Effective Doses with/without the Consideration of the Shielding Factor [mSv].

Computational conditions		TID-14844	RG 1.195
Shielding factor (not considered)	External exposure	1.26	2.01×10^{-2}
	Internal exposure	1.25×10^{2}	9.82×10^{-2}
	Total	1.27×10^{2}	1.18×10^{-1}
Shielding factor (considered)	External exposure	2.62×10^{-1}	4.18×10^{-3}
	Internal exposure	1.25×10^{2}	9.82×10^{-2}
	Total	1.26×10^{2}	1.02×10^{-1}

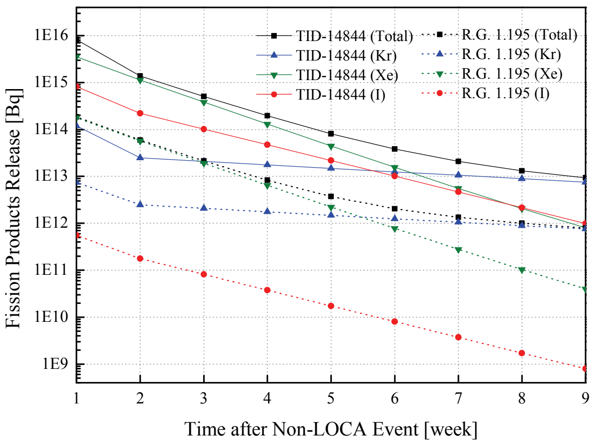

Fig. 3. Fission products released into the environment according to time after the non-LOCA event.

use of shielding factor of domestic building can contribute to the dose assessment for the realistic reflection of normal circumstances. Therefore, the previously-evaluated value of 0.208 for the domestic houses [13] was used in this study. The dose coefficients of the iodine were applied by assuming 91% of the iodine to be elemental, 5% particulate iodine, and 4% organic iodine according to the ratio provided by both regulatory guides.

3. RESULTS AND DISCUSSION

The fission product inventories released into the environment were calculated for a period of nine weeks (approximately 60 days) from the accident point of the fuel assembly meltdown. Figure 3 shows the fission product release individually assessed according to the methodologies of the TID-14844 and the RG 1.195. As a result, their total activities showed 6.74×10^{15} Bq and 2.84×10^{14} Bq, respectively, and the accident source term calculated with the assumption through the TID-14844 were evaluated such that its total amount of noble gases and iodine is 24-times higher than those resulted from the RG 1.195. In light of the nuclides, the difference in noble gases showed about a 20-fold increase. In the case of iodine, the inventory calculated based on the TID-14844 was analysed to be 20,000 times higher owing to the great difference from the release ratio. In particular, applying a decontamination factor of 200 for iodine brings an effect in which the 99.5% the iodine is retained by water.

For an effective dose assessment, the total fission product inventory accumulated for nine weeks was used. The dose calculation was separately carried out regardless of whether the shielding factors are considered for the external exposure or not, and the result of the dose assessment was presented in Table 2.

The external exposure doses applied with the shielding factors were largely decreased in two cases. However, this consequently leads to small differences in the total effective dose, since the external exposure has fewer effects compared with the internal one in terms of the effective dose.

Meanwhile, the doses for each radionuclide derived by the methodologies of the TID-14844 and the RG 1.195 have great differences. These results are caused from the internal exposure by the inhalation of iodine, which is a dominant exposure pathway in contribution of the effective dose. In addition, the iodine's dose coefficient is relatively higher than the others, and especially, the elemental iodine which is assumed to be a significantly large release fraction among them has very high dose coefficients compared with those of organic and particulate iodine. Thus, the doses induced from iodine considerably influence the total effective doses, as shown in Table 3.

4. CONCLUSION

To obtain more conservative results, a radiological dose assessment for a non-LOCA event is occasionally performed following the assumptions of TID-14844, even though there is a regulatory guide of the RG 1.195 with a separate methodology for a non-LOCA. In this case, the assumptions can lead to excessively more severe conclusions than expected, since

Table 3. Effective Doses of Individual Nuclide (SF-considered) [mSv].

Nuclide	TID-14844	RG 1.195	Nuclide		TID-14844	RG 1.195
^{83m}Kr	1.71×10^{-7}	8.56×10^{-9}	^{135}Xe		4.04×10^{-2}	2.03×10^{-3}
^{85}Kr	9.90×10^{-4}	9.90×10^{-5}	^{135m}Xe		1.37×10^{-3}	6.82×10^{-5}
^{85m}Kr	5.13×10^{-3}	2.57×10^{-4}	^{138}Xe		1.49×10^{-2}	7.47×10^{-4}
^{87}Kr	1.61×10^{-2}	8.06×10^{-4}	^{131}I	E*	1.13×10^{-2}	9.02×10^{-2}
				O†	4.13	2.96×10^{-3}
				P‡	2.71	1.83×10^{-3}
88Kr	1.20×10^{-1}	5.98×10^{-3}	^{132}I	E	9.69×10^{-2}	4.83×10^{-5}
				O	5.94×10^{-2}	5.18×10^{-7}
				P	5.90×10^{-2}	3.20×10^{-7}
^{131m}Xe	6.42×10^{-4}	3.21×10^{-5}	^{133}I	E	6.32	3.16×10^{-3}
				O	3.97×10^{-1}	1.05×10^{-4}
				P	3.14×10^{-1}	6.32×10^{-5}
^{133}Xe	1.88×10^{-1}	9.40×10^{-3}	^{134}I	E	5.15×10^{-2}	2.58×10^{-5}
				O	4.05×10^{-2}	8.16×10^{-8}
				P	4.05×10^{-2}	9.18×10^{-8}
^{133m}Xe	2.64×10^{-3}	1.32×10^{-4}	^{135}I	E	5.91×10^{-1}	2.96×10^{-4}
				O	1.71×10^{-1}	7.05×10^{-6}
				P	1.66×10^{-1}	4.15×10^{-6}

* E: elemental, † O: organic, ‡ P: particulate

the TID-14844 does not propose an additional methodology for evaluating a non-LOCA event. In this study, the fission product inventory by a channel blockage accident, which is a non-LOCA event, was evaluated to compare and verify the difference of the results produced based on two regulatory guides. Using two kinds of fission product inventories, the effective doses in the average environmental/meteorological condition were analyzed at the shortest representative distance of 1,000 m where the residents are actually able to live from nuclear reactor facility. As a result, the total activity assessed according to the methodologies of the TID-14844 and the RG 1.195 showed 6.74×10^{15} Bq and 2.84×10^{14} Bq, respectively. In addition, doses for each radionuclide have great differences, since the internal exposure by the inhalation of iodine is a dominant exposure pathway in contribution of the effective dose. Consequentially, it was found that the assumption of the TID-14844 can lead to excessively conservative results for a non-LOCA assessment. Therefore, the differences in the results derived from two guidelines need to be verified before the regulatory authorities or the licensees perform a radiological assessment of the facility for a non-LOCA event.

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