Rational Radiation Safety Technology

Background and Objective

A principle of radiological protection for members of the public is to reduce their exposure due to man-made radionuclides as low as reasonably achievable even in the low-dose region below 1 $mSv.y^{-1}$ (about 0.11 μSv per hour), that is the dose limit for public. This leads to a radiological safety system to require compliance of various safety standards in the extremely low-dose region below 1 $mSv.y^{-1}$. On the other hand, in the field of radiation safety standards in transport of radioactive materials, transport of radioactive waste in a large size container is expected to be realized going with a lot of decommissioning of nuclear reactors.

In this project, reasonable concept and methodology to judge compliance of radiation safety standards is proposed through enhancement of techniques of radiation measurement and safety assessment.

Main results

1. Risk due to man-made radioactive nuclide based on background (BG) cancer risk in Japan

Probability distribution of BG cancer risk (already-existing risk) based on a concept of the International Commission on Radiological Protection (ICRP) and maximum assumed risk distribution due to man-made radioactive nuclides assuming a dose limit for public (the very highest risk distribution when dose limit is complied with) were estimated using Japanese mortality data by prefecture (Fig. 1). As a result, it became clear that more than 98% of assumed risk distribution due to radioactive nuclides is smaller than standard deviation of BG cancer risk distribution that is equivalent to 1 σ. This result revealed that the risk for members of the public is sufficiently small and lower than the risk that is within the deviation of the BG cancer risk, if there is appropriate radiation protection compliance with the dose limit of 1 $mSv.y^{-1}$.

2. Development of the activity distribution estimation system for transport of large radioactive waste

Large radioactive waste, which is generated from the decommissioning of a nuclear power plant, can be transported within the manner of the usual radioactive waste transport procedure when it is demonstrated that its specific activity (activity concentration) is distributed throughout. Thus, we have developed the activity distribution estimation system which utilizes photogrammetry, Monte Carlo calculation and gamma-ray measurement techniques, sequentially (Fig. 2).

To clarify the system ability, accuracy of the BG count rate estimation was evaluated experimentally using mock large radioactive waste and radiation standard sources. Moreover, the relationship between the waste packaging conditions, radiation measurement conditions and the uncertainty in the specific activity were estimated as a function of the filling rate. As a result, it was revealed that the system can give the BG count rate within accuracy of ±30%, and that the system is a practical method since it can confirm the transport requirement with a few gamma-ray measurements. The effect of the system on the manufacturing process of a large radioactive waste package was also estimated. Regarding the typical radioactive waste whose filling rate is 11%, it was indicated that the time increase in the radioactive waste manufacturing process is approximately 10% (Table 1) [L10008].

Other reports Jpn. J. Health Phys., 45 (2), 172-176 (2010).

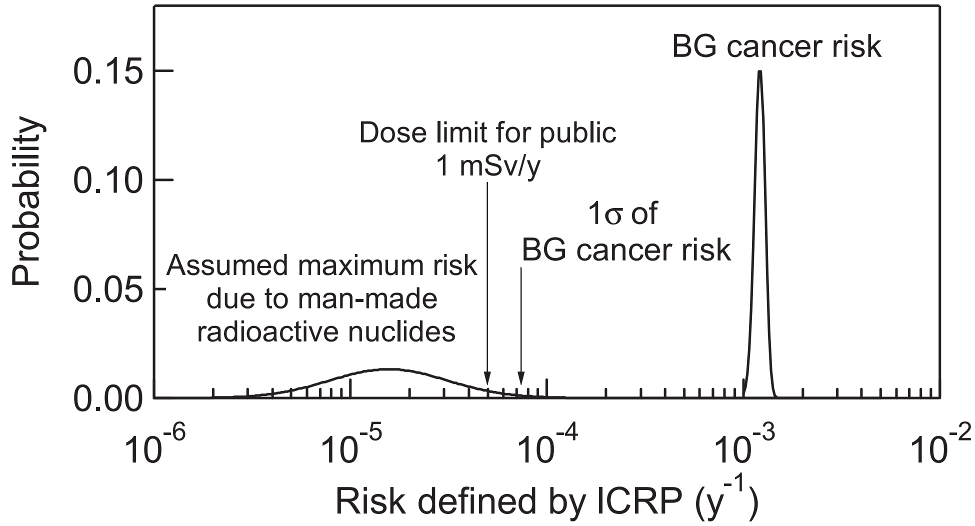

Fig. 1 Comparison between BG cancer risk in Japan and maximum assumed risk distribution due to man-made radioactive nuclides

Probability distribution of BG cancer risk can be considered as a normal distribution. The arithmetic mean is 1.2×10^{-3} y^{-1} and its standard deviation is 7.4×10^{-5} y^{-1}. Risks equivalent to 1 $mSv.y^{-1}$ (dose limit for public) and the maximum assumed risk distribution due to man-made radioactive nuclides are smaller than the risk equivalent to 1 σ of BG cancer risk.

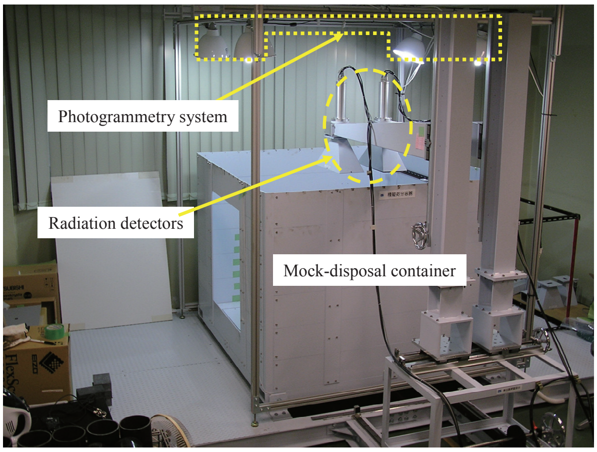

Fig. 2 The activity distribution estimation system for a large radioactive waste package

Table 1 Time increase in the radioactive waste package manufacturing process

Waste placing process (container size: 1.6m x 1.6m)			Specific activity distribution estimation		
Unit placement weight (ton)*1	Filling rate (%)	Placing process (h)	Number of Measure ments	Measure ment time (h)	Increase ratio of the manufacturing time(%)
0.1	5	2.5	2	0.33	14%
	10	5.0	3	0.50	10%
	20	10.0	6	1.00	10%

*1 When the radioactive waste is placed in the disposal container, the use of remote-control method is projected since its activity level is comparatively high.
Possible treatment weight of a manipulator was assumed to be 0.1 ton (100kg) in this study.

In the case that the specific activity of radioactive waste is measured, the accuracy of the specific activity deteriorates because of the increase of the uncertainty in the shielding effect, which is caused by the unknown radiation source position. Thus, in this system, gamma-ray measurement is carried out after the placement of a certain amount of radioactive waste to suppress the uncertainty. For this, in the manufacturing process, although the placement and the measurement are alternately repeated, the effect of the measuring time on the manufacturing process is quite small.